ENCLOSURE

10 CFR 50.59 SUMMARY REPORT

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Summary of Safety Evaluations

SA-SE Number *WBLMGR-97-005-0*

Affected Documents	*Document Type*	*Safety Assessment Title*	*Implementation Date:*
TS Bases LCO 3.4.15 Bases Revision 12	Tech Spec Bases	Heat Tracing for Sample Lines Supplying Monitors	9/10/97

Description of Change, Test, or Experiments

NRC's Notice of Violation (NOV) 390/97-04-08 documented that a procedure for the preventative maintenance (PM) of the heat tracing for Radiation Monitor (RE) 90-129 did not exist. As part of this violation, the impact of heat tracing on the operability of radiation monitors was assessed. Based on this, a commitment was made in the response to NOV 390/97-04-08 to clarify the impact heat trace has on monitor operability. One Technical Specification monitor, 1-RE-90-106, Containment Lower Compartment Monitor, for LCO 3.4.15, Reactor Coolant System (RCS) Leakage Detection also requires heat trace. The WBN design criteria states that monitor 1-RE-90-112, Containment Upper Compartment Monitor shall have the backup capability to monitor the radioactivity in the containment lower compartment atmosphere. Therefore, both of these monitors are considered equal related to this commitment. The implementation of the commitment requires that the Technical Specification Bases for LCO 3.4.15 be revised to define that the sample lines supplying the monitors are heat traced and that during periods when the heat tracing is inoperable, the particulate channel of the monitors will be inoperable. The Bases revision also defines that when the heat tracing is inoperable, grab samples for particulates may not be taken using the sample lines. The alternative compensatory measure of performing an RCS water inventory balance is not affected by this change.

Safety Evaluation Summary

The change required to be made to the Technical Specification Bases does not result from a design change, a test or an experiment and does not result in any physical changes to the plant. Compensatory measures are currently defined in Technical Specification Section 3.4.13, "RCS Operational Leakage." Surveillance requirements (SR) 3.4.13.1 to perform an RCS water inventory balance is required to be implemented during periods when the heat tracing is inoperable and grab samples using the monitor sample lines can not be taken. The Bases revision clarifies that the heat tracing is required to be operable for the monitors to be considered operable. Prior to this clarification, the heat tracing may not have been considered in determining the operability of the monitors. This Bases change is a clarification that ensures conservatism in operability determination. Consequently, an unreviewed safety question does not exist because the probability or consequences of an accident or equipment malfunction is not increased by the change. The change does not create the possibility of an accident or malfunction of a different type than any previously evaluated in the FSAR and the margin of safety defined in the Technical Specification Bases 3.4.15 is not reduced.

SA-SE Number *WBLMGR-97-006-0*

Affected Documents

TRM Change Pkg. No. 97-009

TRM Bases 3.7.3

TRM Revision 4

Document Type

Technical Requirements Manual and Bases

Safety Assessment Title

Timeframe for Conducting Initial Snubber Visual Inspection Prior to Completion of the First Refueling Outage

Implementation Date:

8/18/97

Description of Change, Test, or Experiments

The proposed change revises the BASES for the WBN Technical Requirements Manual (TRM), Section 3.7.3, "Snubbers.. Specifically, the BASES for Technical Surveillance Requirement (TSR) 3.7.3.1 is revised to clarify that the intent of the timeframe for conducting the initial snubber inservice inspection visual examination is prior to completion of the first refueling outage, not "during the refueling outage" as presently stated. This will allow visual inspection of snubbers to begin prior to the first refueling outage (online pre-outage work).

TSR 3.7.3.1 requires that each snubber subject to the requirements of TR 3.7.3 be visually inspected in accordance with the acceptance criteria provided in Table 3.7.3-1 of the TRM. The FREQUENCY for TSR 3.7.3.1 is stated as "In accordance with Table 3.7.3-2" which defines subsequent inspection intervals based on the results of prior inspections, but does not define the timeframe for the initial visual inspection. The BASES for TSR 3.7.3.1 currently states:

"The initial inservice inspection must be Performed on the snubbers during the first refueling outage."

The proposed change to the BASES would be reworded as follows:

"The initial inservice inspection must be performed on the snubbers prior to completion of the first refueling outage."

This will allow inspections to be performed during plant operations (online pre-outage work) in the accessible areas of the plant, i.e., auxiliary building, valve rooms, intake pumping station, etc.

The proposed TRM clarification to allow online visual exams is consistent with the intent of the WBN licensing basis as reflected in References 4 and 5, and the BASES for TR 3.7.3 Technical Surveillance Requirements (page B 3.7-14).

In Reference 4, and in the BASES for TSR 3.7.3, TVA describes the WBN augmented snubber inservice inspection program (as implemented by TR 3.7.3) and requests relief from the 1989 ASME Section XI Code requirements (IWF-5300) related to inservice examinations and tests for snubbers. As discussed in Reference 4, the Augmented Inspection Program, as defined by TR 3.7.3, provides for a level of quality and safety equivalent to or greater than that of the Section XI requirements. The augmented inspection program of TR 3.7.3 is based on the requirements of Generic Letter 90-09 (Reference 8). However, GL 90-09 does not prescribe or define the timeframe for the initial snubber visual examination. Therefore, to confirm the proposed TRM clarification related to the initial snubber visual examination is equivalent to or greater than Code requirements, the 1989 ASME Section XI requirements for snubber inservice examinations were reviewed. The 1989 Section XI (IWF-5300) requirements specify that inservice examinations and tests for snubbers be performed in accordance with the 1988 Addenda to ASME/ANSI Operational and Maintenance (OM)-1987, Part 4. The requirement for the initial visual examination is stated in Part 4, Section 2.3.2.1 (Initial Examination) as:

Safety Evaluation Summary

The proposed change clarifies the timeframe during which the initial snubber augmented ISI visual examinations may be performed and is consistent with the intent of the WBN licensing basis and ASME Section XI ISI requirements (1989 Edition). The proposed change does not alter the design or physical condition of plant components, structures, or systems. Since the proposed change allows performance of the visual examination while maintaining an OPERABLE snubber, the operability of the supported equipment is maintained thus ensuring safety-related equipment remains available to perform its intended function. The design function of the snubber to limit displacement of piping and equipment during severe transients and seismic events and to allow thermal growth is not reduced. In the event INOPERABLE snubbers are identified during the online inspection, the current TR 3.7.3 specifies REQUIRED ACTIONS which must be taken, as would be required for any snubber inoperability identified at power. Therefore, the probability and consequences for malfunctions and accidents previously evaluated in the WBN SAR is not increased, nor is the probability increased for new accidents or equipment malfunctions not previously evaluated. Further, performance of the first visual inspection while at power does not affect postulated radiological releases, or result in anticipated occupational radiation exposures which exceed TVA limits or challenge the regulatory limits of 10 CFR 20 and 10 CFR 100.

Therefore, the proposed change (TR-97-009) to the WBN TRM is acceptable from a nuclear safety standpoint and no unreviewed safety question exists.

"The initial inservice examination of all snubbers shall be initiated at least 2 months after attaining 5% power operation and shall be completed prior to 12 calendar months after attaining 5% power operation."

Thus, the Section XI requirements clearly allowed performance of the initial visual snubber inspection during power operations while ensuring the inspection was completed during the first fuel cycle of operations, e.g., approximately 12 months. The WBN current TRM requirements for snubber inspections are intended to accomplish the same objective, i.e., completion of the first visual inspection of snubbers prior to or during the first refueling outage. With the approval of the Section XI relief request (References 4 and 5), the WBN TRM effectively allows the 12 month criteria of ASME Section XI to be exceeded, provided completion of the initial examination occurs no later than completion of the first refueling outage. The proposed change to the TRM Bases clarifies this intent. In addition, TR 3.7.3 recognizes that some snubbers may be accessible during power operation and allows grouping of the snubbers (accessible/inaccessible) to facilitate the visual inspections. Also, in the event of unsatisfactory snubber visual inspections, the current TRM specifies shortened inspection intervals for subsequent inspections which would necessitate performance of online visual examinations.

Furthermore, the proposal is consistent with the NRC Westinghouse Standard Technical Specification (NUREG 0452, Revision 5, non-MERITS) which prescribed an online visual snubber examination for the first inservice inspection period. In Reference 11, the NRC staff issued this document to Comanche Peak Unit 1. In Section 4.7.9.b of that document (Snubber Surveillance Requirements - Visual Inspections), the first inservice visual inspection of each type of snubber is required to be performed "after 4 months but within 10 months of commencing power operation." This standard Tech Spec also recognizes the grouping of accessible and inaccessible snubbers to facilitate inspection. (NOTE: - This industry standard Tech Spec was in use prior to issuance of Generic Letter 90-09 which allowed licensees to adopt the alternative snubber visual examination program discussed previously.)

Neither the WBN SER and supplements, the WBN FSAR, or the Technical Specifications address performance of inservice examination of snubbers. Therefore, the proposed change to TSR 3.7.3.1 remains consistent with the WBN licensing basis.

SA-SE Number *WBOCEM-96-020-0*

Affected Documents	*Document Type*	*Safety Assessment Title*	*Implementation Date:*
PAI-4.01 R4 CN-5	Procedure	Offsite Dose Calculation Manual Changes	8/1/96

Description of Change, Test, or Experiments

Revision 4, Change Notice 5 makes three changes to WBN PAI 4.01, the Offsite Dose Calculation Manual. The first change is to combine the operability requirements for the iodine and particulate samplers and the sampler flow measurement devices in Table 1.1-2. Since the operability and ACTION statements in this table were identical for these items, combining them into one item on the table is a non-intent change and will not be evaluated further. The second change adds operability clarifications for the isokinetic flow control equipment on the Unit 1 and 2 Shield Building Exhausts (SBE) and the Auxiliary Building Exhaust (ABE), and adds associated ACTIONs to be taken if this equipment is inoperable. The third change made to the ODCM is to raise the allowable monitor tolerance factor for liquid effluent monitor setpoint determinations from 1.5 to 2.0 This change is being made to the ODCM, which is defined as part of the SAR, therefore the change requires a safety evaluation to verify that there is not an unreviewed safety question.

Safety Evaluation Summary

The addition of the operability clarifications and associated actions will not change the ability of the equipment/monitors to perform their design function. The proposed change in the setpoint determination will also not affect the ability of these monitors to perform their design function (to limit radioactive effluent releases to levels below those specified in 10 CFR Part 20). None of the affected equipment/monitors are safety-related, none play a role in the failure modes considered in the SAR accidents or malfunctions, nor do they play any part in the mitigation of any SAR accidents or malfunctions. This change does not affect in any way the accuracy or reliability of effluent, dose, or setpoint calculations. Based on these factors, it can be concluded that these changes do not constitute an unreviewed safety question.

SA-SE Number *WBOCEM-97-001-0*

Affected Documents	*Document Type*	*Safety Assessment Title*	*Implementation Date:*
PAI-4.01 R5 ODCM	Procedure	Incorporate CN-1 and CN-3	5/10/97

Description of Change, Test, or Experiments

ODCM Revision 5.0 incorporates CN-1, CN-2, part of CN-3, and CN4 through CON. Changes made in CN-2, CN4, CN-5 and CN-6 were addressed in previous safety assessments or evaluations performed as a part of those revisions. Changes made as a part of CN-1 and CN-3 being incorporated into the ODCM as a part of this revision will be addressed by this safety evaluation.

The changes made in CN-1 reflect a revision to 10 CFR Part 20: the definition of a member of the public is revised and references to the Controlled Area are changed to reference the Controlled and Restricted Areas. These definition changes implement an NRC regulation and do not affect the intent of the requirements or methodology of the ODCM Controls or calculations, and do not require further evaluation.

CN-3 changes removed radiation monitor unique identifiers from Table 1.1-2 and added applicability notation (5) to Items 2.b, 3.b, and 4.b to clarify that the operability requirement applies only to the sampler, and not to any associated iodine/particulate detection channels. The instrument listed was the ID for the iodine detection channel and was misleading. This is a not a change, rather it is a clarification of the requirement and requires no further evaluation.

On Table 1.1-1, changed the flow indicator IDs for Steam Generator Blowdown to allow use of different instruments than those specified previously. This change substitutes one set of flow indicators for another, has no effect on the Controls or methodologies described in the ODCM; therefore, it requires no further evaluation.

On Tables 1. 1-1 and 2.1-1, clarified the requirements for independent effluent sampling during inoperable monitor periods. The clarification is that two separate individuals will be required to obtain the samples when batch releases are made (from the Liquid Radwaste, Condensate Demineralizer, or Waste Gas Decay System Tanks). This change has no effect on the Controls or methodologies described in the ODCM; therefore, it requires no further evaluation.

The wording in Section 5.3 which describes the change process to be used for the ODCM is revised to specifically disallow the use of non-intent changes which could lead to a violation of the TS and ODCM requirements for evaluation of ODCM changes. This change is administrative in nature and ensures that TS requirements will be met, therefore it requires no further evaluation.

The following are clarifications or corrections of typographical errors and require no further evaluation:
Flow rates given in Section 6.0 for liquid release systems are specified to be design maximums.
Setpoints are specified to apply only to gamma-emitting releases.
Changes to the gaseous release point description.
Typographical errors corrected on Tables 6.2, 6.4, 6.5 and 7.7; in the equations for tritium dose factors in
Section 7.8; and in the minimum building cross-sectional area in Section 7.9.2.

On Tables 1.1-2 and 2.1-2: replaced the flow rate indicator for WGDT with pressure indicators and

Safety Evaluation Summary

The switch from flow rate to pressure indicating instruments to quantify the flow rate for WGDT releases will not change the ability of the Waste Gas Decay System to perform its design function (to keep gaseous effluent releases ALARA by allowing decay of short-lived radionuclides). The new instruments are compliance instruments and will have the same requirements for channel checks, channel calibration, and channel operational tests as the flow rate indicator. In addition, the pressure indicators will provide a more accurate indication of the flow rate since the make-up of the gaseous mix will not factor into the determination as it had using the flow indicator. The proposed change does not affect the ability of the WGDT or Shield Building radiation monitors to perform their design function (to limit radioactive effluent releases to levels below those specified in 10 CFR Part 20). None of the affected equipment/monitors are safety-related, none play a role in the failure modes considered in the SAR accidents or malfunctions, nor do they play any part in the mitigation of any SAR accidents or malfunctions. This change does not affect in any way the accuracy or reliability of effluent, dose, or setpoint calculations. Based on these factors, it can be concluded that these changes do not constitute an unreviewed safety question.

added Action J to Table 1.1-2. to specify the use of pressure indicators to verify flow rates. Also clarified the instrument ID for the source for sampler flow rate information for the Shield Building Exhausts. The latter change substitutes one indicator for another of the same data point, has no effect on the Controls or methodologies described in the ODCM; therefore, it requires no further evaluation. Revised the calibration frequency for the Shield Building Effluent Flow Rate Measuring Device from 12 to 18 months. This revision is made to reflect the requirements of Setpoint and Scaling Document for Instrument Loop No. 1-F-90-400A,B,C,D,-452 and Setpoint and Scaling Document for Instrument Loop No. 2-FI-90400C,452. Because this change makes the ODCM requirements consistent with NUREG-1301, it does not impact the level of control required for effluents and no further evaluation is required.

SA-SE Number *WBOCEM-97-003-0*

Affected Documents	*Document Type*	*Safety Assessment Title*	*Implementation Date:*
PAI-15.01 R3	Procedure	Explosive Gas and Storage Tank Radioactivity Monitoring Program	1/27/97

Description of Change, Test, or Experiments

This change, PAI 15.01 Revision 3, revises the Explosive Gas and Storage Tack Radioactivity Monitoring Program to include discussion of operation of the automatic sequential gas analyzer consistent with FSAR Section 11.3.2. Additionally, this change clarifies sampling requirements for operable and numerable monitor conditions. This change also incorporates discussion of alarms and associated operator actions as described in FSAR Section 11.3.2.

Safety Evaluation Summary

The evaluation of PAI 15.01 Revision 3 indicates that this revision :

a. Implements automatic sequential gas analyzer operation in accordance with FSAR Section 11.3.2.

b. Describes alarms and operator actions in accordance with FSAR Section 11.3.2.

c. Clarifies sampling requirements for inoperable monitors to ensure sampling and analysis consistent with FSAR Section 11.3.2.

The Changes to the Watts Bar Explosive Gas and Storage Tank Radioactivity Monitoring Program detailed by PAI 15.01 Revision 3 are to align the monitoring program with FSAR Section 11.3.2 and related correspondence (SER Supplement 8, SER Supplement 16 and other TVA/NRC correspondence regarding FSAR 11.3). These changes result in additional monitoring not previously required by PAI 15.01 Revision 2. This additional monitoring is in accordance with that described in FSAR Section 11.3.2 and consequently does not create an unreviewed safety question.

SA-SE Number *WBOCEM-97-020-0*

Affected Documents	*Document Type*	*Safety Assessment Title*	*Implementation Date:*
PAI 4.01 R6	Procedure	Condenser Vacuum Exhaust	9/30/97

Description of Change, Test, or Experiments

The applicability note (3) for operability of 1-RE-90-119 and 1-RE-90-129 given in Table 1.1-2 is changed to state that this monitor and sampler are not required to be operable until the normal operational condenser vacuum is established. This revision was addressed in a separate Safety Evaluation (WBPLMN-97-044-0).

The definition of Unrestricted Area/Unrestricted Area Boundary is revised in Section 3.17 of the ODCM. The sentence stating "THE WBN UNRESTRICTED AREA BOUNDARY is equivalent to the SITE BOUNDARY shown in Figure 3.1." is deleted from the definition. This revision is being made due to the proposed leasing of areas within the Site Boundary for commercial use (hay harvest). Calculations performed to support the contracting process for this land use indicated that the maximum resulting dose for an individual resulting from the hay harvesting process would be 12.75 mrem in one year. The ODCM site boundary dose analysis (which considers all available pathways) lists the maximum dose to a member of the public from current methodologies as 25 mrem. The current locations will remain more conservative for use in determining compliance with the ODCM dose and setpoint limits; therefore, the methodologies used to determine the setpoints, doses, and dose projections will not change as a result of this revision.

Clarified the operability requirements and required compensatory actions in Table 1.1-2 for the radiation detector (1,2-RE-90-400) and flow instrument (1,2-FL-90-400) for the shield Building exhausts. This revision does not change the intended compensatory actions or operability requirements for these monitors, it clarifies the intent of the action. Therefore, this change will not be evaluated further.

Changed the operability requirement for Waste Gas Decay Tank (WBDT) release instrumentation in Table 1.1-2 from "at all times" to "during periods of release." This change affects radiation monitor 0-RE-90-118B and pressure indicators 0-PIS-77-115, 0-PIS-77-114, 0-PIS-77-113, 0-PIS-77-100, 0-PIS-77-101, 0-PIS-77-102, 0-PIS-77-145, 0-PIS-77-146, and 0-PIS-77-146. Additionally, the description of the setpoint for 0-RE-90-118 for periods of no release in Section 7.0 is deleted. Releases from these tanks are made as batch releases for which the radiation monitor and pressure indicators provide the assurance that activity and release rate evaluated on the release permit are met during the release. This function will continue to be provided during releases from a WGDT with this revision. This revision makes the monitoring requirements for these batch gaseous release points consistent with the requirements for batch liquid release point monitoring requirements. There is a downstream noble gas monitor (1-RE-90-400) which provides additional assurance that ODCM limits are not exceeded, and this downstream monitor is required to be operable at all times. Because this change for the 0-RE-90-118 and associated WGDT pressure indicators will impact only periods when no releases are being made, it will not lessen the amount of effluent control required by 10 CFR Part 20, Part 50 Appendix I and 40 CFR Part 190. This change will not affect does, dose rate, or dose projection calculations. Setpoint calculations will be affected by this change, in that setpoints for periods when there is no release will no longer be determined; however, the reliability and accuracy of setpoint calculations for WGDT releases will not be affected by this change.

Safety Evaluation Summary

None of the affect equipment/monitors are safety-related, none play a role in the failure modes considered in the SAR accidents or malfunctions, nor do they play any part in the mitigation of any SAR accidents or malfunctions. This change does not affect in any way the accuracy or reliability of effluent, dose, or setpoint calculations. This change does not lessen the level of control of effluents as specified in the Technical Specifications because the revised definition remains consistent with the definitions given in 10 CFR Part 20 and NUREG 1301. Although it may place additional considerations on the selection of the member of the public for whom doses are calculated and reported, it will not affect the methodology used for the calculation of setpoints, doses, or dose projections. These are currently calculated at locations defined in the ODCM at the site boundary. The calculations assume that the individual located there is exposed to all exposure and dose pathways for the maximum time periods specified in Regulatory Guide 1.109. Areas inside the site boundary locations, which will result in higher concentrations; however, the individuals would be at these closer locations for only a fraction of the year, and the dose pathways that will exist at these locations are restricted to milk/beef ingestion from cows fed exclusively the hay harvested from the site, inhalation, and external exposure. These two factors will result in lower overall dose to any member of the public involved in the hay harvest. Based on these factors, it can be concluded that these changes do not constitute an unreviewed safety question.

Added operability requirements to Table 1.1-2 and surveillance requirements to Table 2.1-2 for heat trace on iodine/particulate sample lines on the Condenser vacuum exhaust and both units' Shield Building exhaust. The operability requirements for the heat trace are the same as the requirements for the iodine/particulate sampler. The compensatory actions for loss of the heat trace on the CVE will be to evaluate and estimate activity being released from the CVE using weekly steam generator samples. Compensatory actions for loss of the heat trace on the Shield Building will be to suspend all planned releases from that discharge point until the heat trace is made operable. This change will not lessen the amount of effluent control because compensatory actions are provided to ensure that any releases made will be evaluated and quantified. This change will not impact dose, dose rate, dose projection, or setpoint calculation methodologies.

This revision raises allowable values for the monitor tolerance factor used in the determination of the expected response setpoint in gaseous setpoint caclulations from 1.5 to 2.0. This change is being made to lessen the occurrence of alarms which occur during WGDT releases, and to make the factors consistent with those being used for liquid batch radiation monitors. This change will not lessen the amount of effluent control because it affects only the expected response setpoint. A setpoint is also calculated for each release (the maximum setpoint) which will ensure that the ODCM dose rate limits are not exceeded during the release; this setpoint methodology is not affected by this change. If the expected response setpoint is lower than the maximum setpoint, then it is used as the setpoint for the release. Thus, while this change will affect the setpoint used for 0-RE-90-118, the setpoint determination remains very conservative as compared to the limits. This change will not impact dose, dose rate or dose projection calculation methodologies.

SA-SE Number *WBOENV-97-001-0*

Affected Documents	*Document Type*	*Safety Assessment Title*	*Implementation Date:*
RCI-125 Rev. 4	Procedure	Alternate Valve Options for Flow Control Through the Mobile Demineralizers	1/9/98

Description of Change, Test, or Experiments

The change is alternate valve options for flow control through the mobile demineralizers, and the utilization of the system influent flow path for the system leak test instead of the system effluent. The use of WS-2 as a throttle valve allows for improved flow control and throttling of system flow rate and system pressure control. The use of the influent lines as the leak test flow path eliminates backflow through the vessels and, it reduces the potential for inadvertent fluffing of the resin beds. The Waste Disposal System design or functional requirements presented in the FSAR are not impacted.

Safety Evaluation Summary

This activity is safe because the radwaste system will continue to operate as designed and within the normal parameters. The valve alignment changes are minor. They do not affect the system operation or performance. There is no effect on safety equipment required for safe shutdown of the plant. There are no changes to the system operation, release criteria, or effluent monitoring criteria. All releases will meet the criteria of IO CFR 20 and 10 CFR 100.

SA-SE Number *WBOOPS-96-210-0*

Affected Documents	*Document Type*	*Safety Assessment Title*	*Implementation Date:*
SOI-77.01 Rev. 17 (Effective 8/12/97) SOI-15.01 Rev. 9 (Effective 8/6/97) SOI-14.03 Rev. 10 (Effective 9/10/97)	Procedure	SOIs Revised to Install/Remove Jumpers in Circuity of 0-LPF-27-98 If Inoperative	

Description of Change, Test, or Experiments

This change allows a jumper to be installed in an inoperable Cooling Tower Blowdown (CTBD) Flow Monitor circuitry to enable liquid release valves from the Liquid Radwaste System, Steam Generator Blowdown System and Condensate Polisher Demineralizer Systems to be operable. This change has been evaluated as part of the ODCM the methodology for estimating flow has been addressed in the WBN ODCM. There is no FSAR impact from this change.

Safety Evaluation Summary

The basis for the conclusions of the SE is that the technical specifications and the ODCM recognize the potential inoperability of the discharge flow monitor. They provide provisions for estimating Cooling Tower Blowdown flow when performing radioactive releases with the discharge flow monitor out of service.

SA-SE Number *WBOOPS-96-284-0*

Affected Documents	*Document Type*	*Safety Assessment Title*	*Implementation Date:*
TS Bases Chg. Pkg. 96-015 TS Bases B3.6.3 Revision 8	Tech Spec Bases	TS Bases Clarification	12/6/96

Description of Change, Test, or Experiments

This change to the Technical Specification Bases (TS Bases) clarifies the ACTIONS under TS Bases 3.6.3, "Containment Isolation Valves," with regard to the "administrative controls" that may be utilized for intermittent un-isolation of penetration flow paths using containment isolation valves whose controls are located in the main control room (MCR).

WBN Technical Specification 3.6.3 "Containment Isolation Valves" requires containment isolation valves to be OPERABLE and specifies (in part) ACTIONS to isolate containment penetrations flow paths having inoperable containment isolation valves. The flow path requires isolation by use of at least one closed and de-energized automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured. These Tech Spec Actions are modified by Note 1 which provides that penetration flow path(s) may be un-isolated intermittently under administrative controls. These administrative controls are described in the Tech Spec Bases Section 3.6.3, under ACTIONS, first paragraph as follows:

"These administrative controls consist of stationing a dedicated operator at the valve controls, who is in continuous communication with the control room. In this way, the penetration can be rapidly isolated when a need for containment isolation is indicated."

Tech Spec Bases Change Package No. 96-015 clarifies that for valve controls located in the control room, the dedicated operator may monitor containment isolation signal status rather than be stationed at the valve controls, provided this is a primary responsibility; the dedicated operator may perform other secondary responsibilities which do not prevent adequate monitoring of containment isolation signal status.

SAR IMPACT

This SE is being performed because the Tech Spec Bases (a constituent of the SAR) is being revised and therefore a 10 CFR 50.59 evaluation is required in accordance with Tech Spec requirement 5.6.2, "Technical Specification Control Program." The change does not require a change to the SAR and no adverse impact to the SAR is created.

Safety Evaluation Summary

This change does not affect an FSAR accident or transient analysis. The proposed change does not change the method of assuring containment integrity and does not increase the time to complete containment isolation in the event of an accident. Therefore the proposed change has no adverse impact on the ability of the containment isolation valves to perform their intended function in the event of an accident requiring containment isolation. The change is expected to provide a slight decrease in manual containment isolation completion time because it provides for a single dedicated operator to monitor for the presence of a containment isolation signal and verify and/or perform valve isolation, thereby reducing the time delays associated with coordination/communication between two separate operators to complete this task. Further, since the dedicated operator function under the proposed change does not rely on the required minimum MCR shift crew as defined by Tech Specs and 10 CFR 50.54, the change will have no adverse affect on the mitigation of accidents by Operations personnel.

SA-SE Number *WBOOPS-97-132-0*

Affected Documents	*Document Type*	*Safety Assessment Title*	*Implementation Date:*
DCN CCDCR-62-0 FSAR Change Pkg. 1473 S1	FSAR	Cycle 2 reload - Westinghouse Vantage +/Performance Fuel Design.	9/11/97

Description of Change, Test, or Experiments

The cycle 2 reload fuel will consist of the Westinghouse Vantage +/Performance + (V +/P +) design. The major features of the V +/P + fuel which differ from the cycle 1 recaged Westinghouse Vantage 5H fuel are:

· Zirlo clad fuel rod with reduced internal helium gas prepressurization
· Zirlo fabricated midgrids, thimble tubes and instrumentation tubes
· Integral Fuel Burnable Absorbers (IFBA)
· Wet Annular Burnable Absorbers (WABA)
· Axial blankets consisting of solid pellets enriched to 2.60 w% U-235
· Additional protective bottom grid and associated longer fuel rod end plug for enhanced debris protection
· Numerous material, dimensional, and manufacturing enhancements to the fuel pellets, top and bottom nozzles, and fuel assemble skeleton

The following areas have been assessed during the safety evaluation process: Chemical/mechanical properties, neutronic performance, thermal and hydraulic performance, performance under non-LOCA conditions, and performance under LOCA conditions. These assessments have demonstrated conformance to regulatory requirements and design bases when using V +/P + fuel in the Watts Bar reactor.

Necessary changes to the FSAR text, Tables, and Figures have been identified for Chapters 1, 4, 11, 12, and 15 for utilizing V +/P + fuel in Watts Bar beginning with cycle 2. FSAR changes in Chapters 11, 12, and 15 are contained in FSAR Change Package 1473 and 1488. FSAR changes for Chapter 4 are included in FSAR Change Package 1473 supplement 1 as part of CCDCR-62.

Safety Evaluation Summary

The use of V +/P + fuel in the Watts Bar reactor has been shown to conform to applicable regulatory requirements and remain within the current design bases. This fuel design is compatible with the initial core recaged Vantage 5H fuel, the reactor internals, fuel handling equipment, and refueling equipment. The V +/P + design dimensions are essentially identical to the current fuel from an exterior assembly envelope and reactor internals interface standpoint.

SA-SE Number *WBOOPS-97-143-2*

Affected Documents

WB U1C2 Operation (all modes) With Redesigned Core
FSAR Change Pkg. 1493

Document Type

FSAR

Safety Assessment Title

Reactor Operation in all Modes for Cycle 2 Operation

Implementation Date:

9/30/97

Description of Change, Test, or Experiments

This safety assessment/safety evaluation (SA/SE) considers reactor operation in all modes for cycle 2 operation to a maximum cycle core average burnup of 19,000 MWd/MTU, including a power coastdown. Core load and operation in mode 6 was also previously considered in revision 1 of this SA/SE. The revision O SA/SE is superseded by revisions 1 and 2 due to required modifications to the cycle 2 loading pattern. These revisions were required after one fuel assembly with leaking fuel rod(s), which was scheduled for reuse in cycle 2, was identified. The loading pattern was revised to remove this defective fuel assembly. Confirmation that the redesigned cycle 2 core will remain within the bounds of all FSAR accident analyses is based on the Reload Safety Evaluation performed by Westinghouse.

WBN unit 1 will be refueled by replacing 84 burned fuel assemblies with 84 fresh Westinghouse Vantage +/Performance + (V +/P +) fuel assemblies and shuffling the remaining burned fuel assemblies for cycle 2. Fuel inserts including secondary sources, rod cluster control assemblies (RCCAs), and plugging devices will also be shuffled. Wet Annular Burnable Absorbers (WABA) will be utilized in selected core locations where discrete absorbers are required. In addition, 32 Tritium Producing Burnable Absorbers (TPBAR) in four lead test assemblies will also be loaded.

The current WBN unit 1 cycle 2 Core Operating Limits Report (COLR, revision O) will also be revised to reflect changes, which were required for the revised loading pattern, to the following specific operating limits:

· As-measured MTC limit is changed to -2.1 x 10-5 Dk/k/°F from -2.3 x 10-5 Dk/k/°F.
· New values of W(Z) are provided.

The remainder of the current cycle 2 COLR remains applicable for the redesigned core loading pattern.

SAR Impacts

Necessary changes were identified to FSAR Chapters 1, 4, and 15 associated with the use of the V +/p + fuel and the cycle 2 reanalyzes. These changes are incorporated in FSAR Change Package 1473, 1473 Supplement 1, and 1488.

FSAR changes necessary for insertion of the TPBAR LTAs have been identified to Chapters 4, 11, and 15. These changes are included in FSAR Change Package 1483.

In addition, the cycle 2 reload safety analyses performed by Westinghouse identified one FSAR statement that cannot be met with the cycle 2 core. FSAR Section 9.3.4.3.1 currently states that "the rate of boration, with a single boric acid transfer pump operating, is sufficient to take the reactor from full power operation to 1% shutdown in the hot condition, with no rods inserted, in less than 90 minutes. In less than 100 additional minutes, enough boric acid can be injected via the normal boron charging path to compensate for xenon decay." At the Technical Specification minimum boration

Safety Evaluation Summary

Operation of the redesigned Watts Bar cycle 2 has been demonstrated to conform with all applicable design and performance standards, and ensures that all pertinent licensing basis acceptance criteria are met. Based upon the following:

· The end-of-cycle 1 core average burnup is bounded by 16,244 and 17,398 MWd/MTU (actual EOC burnup was 16940.4 MWd/MTU)
· The cycle 2 core average burnup will not exceed 19,000 MWd/MTU including power coastdown operation; and
· There is adherence to plant operating limitations as given in the Technical Specifications and Technical Requirements.

There are no unreviewed safety questions identified as a result of the redesigned Watts Bar cycle 2 reload core.

rate (35 gpm), 170 minutes are required to fully borate to compensate for xenon decay. The FSAR requirement of 100 minutes will be revised to 200 minutes in FSAR Change Package 1493. Revising this requirement to 200 minutes will provide sufficient margin for future cycles. The change of the value to 200 minutes is acceptable because xenon decay below the initial equilibrium level will not begin until approximately 25 hours after shutdown, as stated in the FSAR.

SA-SE Number *WBORAD-95-008-1*

Affected Documents	*Document Type*	*Safety Assessment Title*	*Implementation Date:*
Procedure No. SOI-7701 WBPER960100	Procedure	Administrative Change to Provide Corrective Action for Adverse Condition WBPER960100	3/29/96

Description of Change, Test, or Experiments

This change is to 0-RCV-77-43. The change provides a means to operate 0-RCV-77-43 if the radiation monitor 0-RM-90-122 is inoperable. Chapter 11 of the SAR indicates that "once the fluids are sampled, they are pumped through a normally locked closed manual valve and a remotely operated control valve, interlocked with a radiation monitor and a flow element in the cooling tower blowdown line." This change has been evaluated as part of the ODCM and is discussed in the WBN ODCM.

Safety Evaluation Summary

The basis for the conclusions of the SE is that the technical specifications and the ODCM recognize the potential inoperability of the radiation monitor. They provide the sampling methodology for performing radioactive releases with the radiation monitor out of service.

SA-SE Number *WBORAD-96-007-0*

Affected Documents	*Document Type*	*Safety Assessment Title*	*Implementation Date:*
FSAR Change Pkg. 1437	FSAR	Radiological Control Program Changes	11/13/96

Description of Change, Test, or Experiments

FSAR section 12.5.2 addresses the use of dosimetry at WBN. As a result of the electronic dosimetry becoming NVLAP accredited, WBN is revising the section of the FSAR to allow for the use of the electronic dosimeter as the legal dose of record. The use of the electronic dosimeter will eliminate the need for primary and secondary monitoring devices. The electronic dosimeter can perform both functions.

The second aspect of the FSAR change is to evaluate the use of the TLD without a secondary monitoring device.

Safety Evaluation Summary

SSER 10 page 12-1 section 12.6 states "personnel radiation monitoring is conducted quarterly using TLD's supplemented by real-time dose tracking with direct reading dosimeters." The SSER goes on to state that the program will "meet the requirement of 10 CFR 20.1501(c)." SSER 14 page 12-2 states that "dosimeters used for monitoring radiation doses will be processed by a laboratory accredited under the National Voluntary Laboratory Accreditation Program as required by 10 CFR 20.1501(c)." This FSAR revision may eliminate the use of TLD's as described in SSER 10. However, the replacement device meets the requirements of 10 CFR 20.1501(c). The replacement device also provides real-time dose tracking as described in SSER 10. Due to the fact that the electronic dosimeter is NVLAP approved and the statements of SSER 14, the change to the electronic dosimeter as the device of record is equivalent to the current program. In both supplements, the basis for approval of the dosimetry program is the compliance with 10 CFR 20.1501(c). This revision complies with the requirements of 10 CFR 20.1501(c). The electronic dosimeter can only detect gamma radiation. Therefore, for mixed radiation fields, mixed radiation field monitoring will be provided per SSP-5.01 "Radiation Protection Plan" which states "RADCON determines the type, placement, and processing frequency of dosimetry based on work conditions and the individual's exposure history." This determination will satisfy the monitoring requirements for mixed radiation fields. The TLD meets the requirements of 10 CFR 20.1501(c). However, the TLD does not provide real-time dose tracking as discussed in SSER 10. There are no regulatory requirements for real-time dose tracking. Additionally, the TLD must be used in conjunction with some other type of monitoring device that satisfies the requirements of TS 5.11.

SA-SE Number *WBOTSS-95-578*

Affected Documents

TACF No. 0-95-010-077

Document Type

Temporary Alteration

Safety Assessment Title

U1/U2 interface boundary relocated to allow repair of an external leak on valve.

Implementation Date:

Description of Change, Test, or Experiments

This change relocates the U1/U2 boundary to allow repair of an external leak on valve 2-ISV-77-574, the current U1/U2 interface point. Valve 2-ISV-77-574 is the isolation point between the Monitor Tank and the U2 PMW Tank and pumps. The change involves moving the boundary toward Unit 1, which means that several boundary points are needed to isolate Unit 1 from Unit 2, as detailed on TACF 0-95-010-077.

Safety Evaluation Summary

This change is safe because the temporary isolation boundaries adequately isolate U1 from U2. Releases from the Monitor Tank are not allowed while this temporary condition is in effect.

SA-SE Number *WBOTSS-96-082-0*

Affected Documents

Disabling Alarm Associated with 0-LS-78-3

Document Type

Temporary Alteration

Safety Assessment Title

Disablity the Alarm Associated with 0-LS-78-3 During the Period of Time the Spent Fuel Pit is Drained for Installation of New Spent Fuel Storage Racks

Implementation Date:

Description of Change, Test, or Experiments

This Safety Evaluation covers the disabling of the level alarm for the spent fuel pit [1-XA-55-6C/128A]. This disabling will be accomplished electronically, through the annunciator system itself. The purpose of disabling the alarm is to prevent window 1-XA-55-6C/128A from being continuously in an alarm status while the water is drained from the pit for the installation of new spent fuel storage racks. The sole impact to the plant of this change is that the spent fuel pit level alarm will not alarm during the period of time the pit is drained for reracking. The alarm must be enabled prior to storage of irradiated fuel assemblies in the spent fuel pit.

Safety Evaluation Summary

As documented in the Safety Assessment, the sole impact associated with this activity that warrants a Safety Evaluation is that the level alarm for the Spent Fuel Pit is described in the FSAR. Disabling this alarm does not significantly alter the function of the system, but does differ from the description provided in the FSAR. The alarm function is intended to provide the operator with a warning that the potential exists for one of two conditions warranting immediate attention. These conditions are the potential to uncover irradiated fuel stored in the spent fuel pit and the potential to increase radiation levels in the spent fuel pit area by reducing the shielding provided by the water above irradiated fuel assemblies. Neither of these conditions are a concern until irradiated fuel is stored in the spent fuel pit. This will not occur until one of the following: the unit reaches the first refueling outage, or an accident occurs prior to the first refueling outage that necessitates moving irradiated fuel from the reactor vessel to the spent fuel pit.

The Technical Specifications potentially associated with this level alarm are not applicable during the period of time the alarm will be disabled. The potentially associated Technical Specifications and Technical Requirements only apply when irradiated fuel assemblies are present in the spent fuel pit or are be moved in the refueling cavity. The spent fuel pit level alarm must be reenabled prior to storing irradiated fuel in the spent fuel pit and prior to moving irradiated fuel in the refueling cavity.

SA-SE Number *WBOTSS-96-084-0*

Affected Documents	*Document Type*	*Safety Assessment Title*	*Implementation Date:*
FSAR Change Pkg. 1412 SOI-59.01 R13, CN-2	FSAR	Demineralized Water from the Ecolochem System to be Transferred Directly to the Demineralized Water Distribution System	2/15/96

Description of Change, Test, or Experiments

The document referenced above is a change to the operating instruction on the demineralized water distribution system which will allow demineralized water from the Ecolochem system to be transferred directly to the demineralized water distribution system. This change will allow demineralized water to be transferred directly to the condensate storage tanks from the Ecolochem system.

The need for this change came about as a result of high dissolved oxygen levels in the condensate storage tanks. The high levels are attributed to exposure to the atmosphere in the 500,000 gallon demineralized water storage tank which is the normal discharge path of the Ecolochem system. Transferring the demineralized water directly to the condensate storage tanks from the Ecolochem system would prevent exposure of the water to the atmosphere and will provide much lower dissolved oxygen levels and greatly reduce the amount of nitrogen required to sparge the condensate storage tanks.

SAR IMPACT
This Safety Evaluation is being performed solely because a change to FSAR section 9.2.3.2 is required to reflect this procedure change.

Safety Evaluation Summary

This procedure change does not affect any FSAR accident or transient analysis. This procedure change utilizes existing system piping and valves and does not place the system in an abnormal mode of operation. This system is not required for the mitigation of any FSAR Chapter 15 Design Basis Accidents and no new accidents or malfunctions are introduced. Based on the evaluation of the effects, it is concluded that this change is acceptable from a nuclear safety standpoint and no unreviewed safety question exists.

SA-SE Number - *WBOTSS-96-247-0*

Affected Documents	*Document Type*	*Safety Assessment Title*	*Implementation Date:*
TACF No. 1-96-33-77	Temporary Alteration	Add temporary pressure gauge outboard to allow RCDT pressure indication.	2/6/97

Description of Change, Test, or Experiments

This change installs a temporary pressure gauge down stream of 1-ISV-77-592 in order to allow monitoring the Reactor Coolant Drain Tank pressure while repairs of the permanent process instrument 1-PT-77-2 are implemented. The existing caps at 1-ISV-77-592 will be removed and replaced with pipe fittings and temporary tubing leading to a connector for a temporary pressure gauge installation.

Although 1-47W830-1 is FSAR fig. 11.2-1, the temporary change to install a pressure gauge outboard of 1-ISV-77-592 is not a permanent installation and therefore will not require an FSAR change. The Waste Disposal System design or functional requirements presented in the FSAR are not impacted.

Safety Evaluation Summary

This change is safe because the piping fittings, tubing and pressure gauge used are rated for design pressure, and valve 1-ISV-77-592 will be maintained closed. This valve will be opened by operations when RCDT pressure readings are required and then closed. The TACF does not affect any FSAR evaluations previously performed. No new accidents or equipment important to safety malfunctions are created. Technical Specification is not affected. Therefore, the TACF is acceptable from a nuclear safety standpoint and no USQ exists.

SA-SE Number *WBOTSS-96-290-0*

Affected Documents	*Document Type*	*Safety Assessment Title*	*Implementation Date:*
FSAR Change Pkg. 1451	FSAR	FSAR Change to Reference Rev. 1 to ASME Code Case N-416	9/23/97

Description of Change, Test, or Experiments

This change revises the FSAR to include reference to Revision 1 of ASME Code Case N-416 [N-416-1]. The original issue of the Code Case is already described in the FSAR for application to Code Class 2 systems. The original issue of the Code Case addressed deferral of the hydrostatic testing required following repairs or replacements by welding on the pressure boundary of Code Class 2 systems. The circumstances specified in the Code Case were that the valves necessary to provide isolation for the test were not present or that it was necessary to secure, gag, or remove safety or relief valves to accomplish the test. When these circumstances existed, the post-repair/replacement hydrostatic test could be deferred until the next regularly scheduled system hydrostatic test. This deferral period could have been as long as 10 years of actual reactor operation.

Revision 1 of Code Case N-416 expands the scope of coverage from Code Class 2 only to Code Classes 1, 2, and 3. It also removes the limitations involving the inability to isolate the test boundary or the need to gag or otherwise secure a safety or relief valve. Additionally, rather than deferring the hydrostatic test for a period of up to 10 years, it replaces the hydrostatic test with a system inservice test performed at nominal operating temperature and pressure. This change involves updating the FSAR to address Code Case N-416-1 and revising SSP-8.05 to implement the provisions of Code Case N-416-1.

In addition to the requirements of the Code Case, an augmented examination requirement is being added when the Code Case is utilized on Code Class 3 piping. For Code Class 1 and 2 piping, the Non-Destructive Examination [NDE] required by the Code provides a very high level of confidence that through wall defects do not exist. However, for Code Class 3 piping, this NDE is not generally required. Therefore, when applying the provisions of this Code Case to Code Class 3 piping, a surface examination will be required of the root pass layer of the weld when the Code dictates that a surface examination is required for the completed weld. As described in subparagraph ND-5222, this NDE is required for piping greater than 2" NPS when performing butt or socket welds.

The NRC has endorsed the original issue of Code Case N-416 for use without restriction in Regulatory Guide 1.147, Inservice Inspection Code Case Acceptability ASME Section XI division 1. Code Case N-416-1 has not been endorsed by the NRC in Regulatory Guide 1.147. However, paragraph (g)(5)(iv) of 10CFR50.55a states that when an examination requirement is determined to be impractical by the licensee and is not included in the inservice inspection program the basis for this determination must be demonstrated to the satisfaction of the NRC not later than 12 months after the expiration of the initial 120-month period of operation from the start of facility commercial operation. NRC has approval of use of the Code Case at WBN in a letter dated September 23, 1997.

Safety Evaluation Summary

The acceptability of performing nominal operating pressure tests in lieu of hydrostatic tests is supported by the recent approval by the American Society of Mechanical Engineers [ASME], Board of Nuclear Codes and Standards, of ASME Code Case N - 16-1, Alternative Pressure Test Requirements for Welded Repairs or Installation of Replacements Items by Welding. This code case allows a system leakage test at nominal operating pressure and temperature to be used in lieu of a hydrostatic test, provided that NDE of the weld(s) is performed in accordance with the methods and acceptance criteria of the applicable Subsection of the 1992 Edition of the ASME Boiler and Pressure Vessel Code, Section III. This guidance is sufficient for Code Class 1 and 2 components since the code requires volumetric examination of repairs or replacement in Code Class 1 and 2 components. However, the code only requires a surface examination of the final weld pass in Code Class 3 components. Accordingly, the proposed change supplements the examination requirements specified in N-416-1 with an additional surface examination of the root pass when Subparagraph ND-5222 requires a surface examination of the final weld.

Piping components are designed for a number of loadings that would be postulated to occur under the various modes of plant operation. Hydrostatic testing only subjects the piping components to a small increase in pressure over the design pressure and, therefore, does not present a significant challenge to pressure boundary integrity. Accordingly, hydrostatic pressure testing is primarily regarded as a means to enhance leakage detection during the examination of components under pressure, rather than solely as a measure to determine the structural integrity of the components.

Industry experience has been that leaks are not being discovered as a result of hydrostatic test pressures propagating a preexisting flaw through wall. Experience indicates that when leaks are found, in most cases they are found when the system is at normal operating pressure. This is largely due to the fact that hydrostatic pressure testing is required only upon installation and then once every 10-year inspection interval, while system leakage tests at nominal operation pressures are conducted a minimum of once each refueling outage for Class 1 systems and each 40-month inspection period for Class 2 and 3 systems. The leaks that are being identified are being found during the more frequent system leakage tests. In addition, leaks may be identified by plant operators during routine system walkdowns.

Following the performance of welding on the pressure boundary, the code requires volumetric examination of repairs or replacements in Code Class 1 and 2 systems, but only requires a surface examination of the final weld pass in Code Class 3 systems. There are no ongoing NDE requirements for Code Class 3 components except for visual examination for leaks in conjunction with the 10-year hydrostatic tests and the periodic leakage tests.

Considering the NDE performed on Code Class 1 and 2 systems and considering that the hydrostatic pressure tests rarely result in pressure boundary leaks that would not occur during system leakage tests, there is very little increased assurance of the integrity of Class 1 and 2 systems added by the hydrostatic test requirement.

However, considering the nature of NDE requirements for Code Class 3 components, eliminating the hydrostatic pressure testing and only performing system pressure testing is not an acceptable alternative to hydrostatic testing unless additional NDE is performed. Accordingly, the additional requirement to perform a surface examination on the root pass layer of butt and socket welds in the pressure retaining boundary of Code Class 3 components when the surface examination method is specified in accordance with Section III is being incorporated. With this provision applied to Code Class 3 components, it is our conclusion that the substitution of system pressure tests at operating pressures and temperatures for the elevated pressure hydrostatic test does not present an unsafe or adverse condition and no USQ exists.

SA-SE Number *WBOTSS-96-307-0*

Affected Documents	*Document Type*	*Safety Assessment Title*	*Implementation Date:*
1-PI-OPS-1-TB TP-06-024	Temporary Procedure	#2 Feedwater Heater's Operating Vent Valves	8/28/97

Description of Change, Test, or Experiments

This temporary procedure provides the steps to determine the length of time the #2 feedwater heater's operating vent valves can remain closed during full power operation without significantly affecting heater performance. It also determines the length of time the subject vent valves should be opened following a closure period to effectively vent any collected non-condensables to the condenser. This determination is necessary since the subject vent lines were determined to not contain flow control orifices. The lack of orifices has contributed to rapid erosion of the vent lines downstream of the vent valves. Operating the #2 FW heaters with the subject operating vent valves normally closed but cycled open periodically to vent non-condensables should reduce the rate of erosion in these lines until the permanent repair / replacement can be made. This temporary procedure is applicable to a single #2 feedwater heater, 1-FWHT-002-C2, and its associated vent valve, 1-VTV-006-682. This instruction uses the plant process computer to monitor heater terminal temperature difference (TTD), and final feedwater temperature, and other process parameters,. During the performance of this instruction, heater 2C's TTD and final feedwater temperature will be monitored to determine when venting is necessary, to determine when venting has adequately been completed, and to determine if feedwater temperature change parameters have been approached. Final feedwater temperature change is one criteria for reopening the subject vent valve.

The SAR, section 10.4.10.2, describes the subject vent valve alignment during normal operation. The SAR states in part, "venting [of the feedwater heaters] to the main condenser during normal operation is accomplished through continuous 'free blowing' orifices...". This temporary procedure will close the subject vent line for a period of time during normal operation; thereby, conflicting with the reference. The purpose of the procedure is to determine the length of time the valve can remain closed and the length of time required to provide adequate venting after the closure period so that adequate venting can be accomplished with normally closed but periodically opened vent valves. The result of either alignment is adequately vented feedwater heaters.

Section I5.2.10 of the SAR addresses excess heat removal events caused by feedwater system malfunctions. Although not specifically addressed in the SAR, the Design Basis Events Design Criteria, WB-DC-40-64, section 4.23.1, defines the event as "secondary system malfunctions which cause excessive feedwater addition or reduction in feedwater temperature. The resultant effort is an unplanned increase in heat removal by the secondary system, accompanied by a decrease in reactor coolant temperature which increases core reactivity and can lead to an increase in core power. Either a control system malfunction or operator error may be the initiating event." In addition, the SER, section 15.2.2, lists "decrease in feedwater temperature" as an event that WBN has analyzed which produced increased primary system cooling. Although WB-DC-40-64 addresses the defined event as caused by excessive feedwater addition in detail, it does not address the defined event as caused by a reduction in feedwater temperature. The Accident Analysis Parameter Checklist (AAPC) offers insight as to why the defined event as caused by a reduction in feedwater temperature was not addressed. Section 4.1.9.1 of the AAPC states in part that the defined event could result from the "failure of a bypass valve that diverts flow around the low pressure feedwater heaters for plants so equipped. Watts Bar Units 1 and 2, however, are not equipped with this type of bypass valve." The referenced section of the SAR is pertinent to the subject temporary procedure since the closing the subject vent line for an extended period of time could lead to air binding of the 2C heater and a

Safety Evaluation Summary

This temporary procedure does not involve an unreviewed safety question. The controlled closure of this valve will not create the possibility of any other type accident or any additional failure modes. The only credible type of accident that closing this vent valve could be associated with is an Excessive Heat Removal Due To Feedwater System Malfunctions accident. Since the overall feedwater temperature change is administratively limited to no more than 1.5°F reduction and the temperature reduction will be periodic, the temperature reduction will have an negligible impact on the licensing basis accidents evaluated previously in the SAR.

The controlled closure of this valve does not increase the probability of occurrence of a malfunction of equipment important to safety. This is a non-quality related valve between two non-quality related components. All required equipment important to safety will remain in-service during this activity.

The controlled closure of this valve will not increase the consequences of an accident previously evaluated. The closure of this valve does not change the reactor coolant system's accident mitigation performance and therefore does not increase the potential for radiation releases to the environment.

Based on these arguments, the change does not involve an unreviewed safety question.

subsequent reduction of heat transfer to the feedwater system. If allowed to progress, final feedwater temperature could be reduced. The potential affect from heater 2C air binding is mitigated by the three downstream #1 heater's ability to draw more extraction steam in response to the cooler inlet water conditions and by the fact that the scope of the procedure is limited to one #2 heater in the three heater strings. In addition, the subject procedure administratively limits the decrease in heater 2C's outlet temperature /TTD to 3°F and the decrease in final feedwater temperature to 1.5°F. Although not explicitly bounded by the SAR or by the Design Criteria, the 1.5°F decrease limit has been determined to not impact the analysis described in the SAR. Specifically, when questioned about the subject activity, Westinghouse stated "Since the overall feedwater temperature should change by no more than 1.5°F and the temperature reduction will be periodic, it is believed that the temperature reduction will have an negligible impact on the licensing basis LBLOCA, SBLOCA, SGTR and Transient Analyses. Furthermore, there should be no impact on the containment integrity analysis nor in the steamline break mass and energy releases. Also, there should be no impact on the TRS/ESFAS setpoints or initial condition uncertainties."

SA-SE Number *WBOTSS-96-314-0*

Affected Documents	*Document Type*	*Safety Assessment Title*	*Implementation Date:*
STI-96-01	Special Test	Primary and Secondary Plant System to Sustain a 50% Load Reduction Without Manual Intervention	10/1/96

Description of Change, Test, or Experiments

This Special Test Instruction will demonstrate the ability of the primary and secondary plant systems, including automatic control systems, to sustain a 50% load reduction without manual intervention.

This test will be initiated from a steady state power level of 95 to 100% after insuring all required control systems are operating normally in automatic. A rapid 50% reduction in Turbine/Generator load will be initiated using the Turbine/Generator EHC system. primary and secondary plant parameters will be monitored throughout the transient until stability is achieved at approximately 45 to 50% power.

A 50% load reduction test was performed during the Power Ascension Test Program under the guidance of 1-PAT-1.3 to verify the acceptance criteria listed in the FSAR Chapter 14, Table 14.2-2, Sheet 35. During this PAT a two step load drop occurred instead of a one step, however all acceptance criteria was determined to have been satisfied. This special test is being performed to satisfy NRC concerns over the adequacy of the PAT. This special test will demonstrate that the FSAR acceptance criteria can be met for a single step load decrease in power.

SAR IMPACT
This Safety Evaluation is being performed because SSP-8.04 requires SA/SE's for all special tests. Also, FSAR Chapter 14.0 describes the 50% load reduction test as being part of the initial start-up test program. These tests were performed prior to declaring the plant in commercial operation. Since the plant is now in commercial operation and this special test is being performed outside of the Power Ascension Test Program a SE is required.

Safety Evaluation Summary

This procedure does not affect any FSAR accident or transient analysis. The 50% load reduction test is described in Chapter 14 of the FSAR. Also, the plant is designed to handle a 50% drop in load without a reactor trip or lifting safeties (10% with control rods and 40% via steam dumps). This test .is also bounded by FSAR Chapter 15 Design Basis Accidents and no new accidents or malfunctions are being introduced. Section 15.2.7 of the FSAR addresses loss of external load and/or turbine trip. This is a Condition II event that assumes a loss of load from 102% rated power. Results show that the plant design is such that a total loss of external electrical load without a direct or immediate reactor trip presents no hazard to the integrity of the RCS or Main Steam System. Based on the evaluation of the effects, it is concluded that this procedure is acceptable from a nuclear safety standpoint and no unreviewed safety question exists.

SA-SE Number *WBOTSS-96-444-0*

Affected Documents	*Document Type*	*Safety Assessment Title*	*Implementation Date:*
Maint. WR/WO No. C373740/9602409000	Work Order	Add temporary temperature gauges at test connection valves in the Nitrogen supply line, waste gas vent header, or gas analyzer line to the Reactor Coolant Drain Tank in order to determine if there is a blockage in the Nitrogen supply line restricting the flow.	12/16/96

Description of Change, Test, or Experiments

This Safety Assessment/Safety Evaluation is required to allow temporary pressure gauges installed outboard of 1-TV-77-481, 1-TV-77-504, 1-TV-77-506, and 1-TV-77-507 on the RCDT vent header, Nitrogen supply and gas sample lines to enable trouble shooting activities to be performed, which will allow identification of potential faulty components installed in these headers that are restricting flow. The installation and configuration control for the pressure gauges will be implemented by WRC373740/WO96020409000. After the gauges are installed and the test valves are open per the WR/WO then the section of SOI-77.01 that purges the RCDT will be performed in order to flow Nitrogen through the line. When Nitrogen flow is present then pressures will be recorded and that data will be used to locate the suspected restriction in the line. Each existing cap will be removed and replaced with fittings and temporary tubing leading to a connection for each pressure gauge. Configuration control of the above test vent valves will be maintained in the WR/WO using a configuration log. The WR/WO will close the test vent valves, remove the pressure gauges and replace the pipe caps at the conclusion of data taking activities.

Although 1-47W830-1 is FSAR Fig.11.2-1, the temporary installed pressure gauges are not permanent installations and therefore will not require an FSAR change. The Waste Disposal System design or functional requirements presented in the FSAR are not impacted.

Safety Evaluation Summary

This activity is safe because the piping fittings, tubing and pressure gauges used will be rated to withstand the expected pressure of the system when aligned for normal RCDT purging configuration. Expected pressure is approximately 6 psig or lower. The associated test vent valves will be opened momentarily to collect pressure readings when Nitrogen flow is present and then closed and the gauges removed from the system. This activity does not affect any FSAR evaluations previously performed. No new accidents or equipment important to safety malfunctions are created. Opening of 1-TV-77-481 is administratively controlled and bounded by the requirements of LCO 3.6.3. Therefore this WR is acceptable from a nuclear safety stand point and no USQ exist.

SA-SE Number *WBOTSS-97-015-4*

Affected Documents	*Document Type*	*Safety Assessment Title*	*Implementation Date:*
DCN W-39476-A	DCN	Flow Rate of ERCW Through Chiller Condenser	11/4/97

Description of Change, Test, or Experiments

TACF 0-97-003-067 permits modification of the Essential Raw Cooling Water (ERCW) Temperature Control Valves (TCVs) associated with the Main Control Room (MCR) chillers and the Electric Board Room (EBR) chillers. These chillers are required to maintain acceptable environmental conditions within the Control Building during any mode of system operation following any single active failure. The EBR chillers are not required during a flood. Each chiller is 100% capacity. The chillers and associated ERCW piping are designed to withstand the safe shutdown earthquake. These ERCW TCVs are not ASME as they have been exempted under ASME Section XI. The active function of these valves for design basis events is to modulate to control MCR and EBR chiller heat load.

The design function of the TCVs is to control the flow rate of ERCW water through the chiller condenser. The ERCW water is intended to remove excess heat from the chiller. When the chiller is not in service (standby), ERCW flow rate must be limited to near zero through the TCVs when ERCW supply temperatures are below 65 degrees F. The TCVs when closed do not provide complete isolation. However, the insignificant flow (near zero) through the TCVs and through the associated tubing is consider negligible and should not affect the chillers. Leakage of ERCW through the standby chiller condenser cold river water temperature conditions causes the chiller oil to become too cold. When the oil is too cold, the chiller can not be assured to start and remain operating when automatically started from a standby condition.

The differential pressure across the TCVs has been measured to be greater than 100 psid. The manufacturer of these TCVs, METREX, has stated that the maximum differential pressure should be no greater than 90 psid. ERCW water leakage can be expected if differential pressures across TCVs is greater than 90 psid. The changes described by TACF may be implemented to permit these valves to control ERCW flow at differential pressures greater than 90 psid.

Presently the operating chillers continue to perform acceptably with the TCVs operating in the as-found condition (leaking). LCO 3.7.11 and TOE 1-96-031-9019 Rev. 2 permits continued plant operation with one train chiller inoperable for up to 30 days.

The TCVs will be modified as follows:

Install stainless steel tubing to connect the low pressure side of the TCV actuator to an ERCW vent valve located upstream of the TCV to permit upstream ERCW header pressure to be sensed on the low pressure side of the TCV actuator.
Tubing has two metering valves in line to adjust the pressure to the low pressure side of the TCV actuator. Valves will be adjusted to a position which will ensure that an acceptable differential pressure is maintained across the TCV actuator (60 to 90 psid). Tubing is type 316 stainless steel, QA quality level 1. Valves used are globe type valves manufactured by DRAGON. These valves are ASME Section III Code Class 2.
The procurement and installation of this tubing and associated valves is documented as Quality Level I.

Upon installation of tubing and valves the following testing will assure operability of chillers:

Safety Evaluation Summary

This change improves the reliability of the ERCW TCVs to isolate ERCW flow to the MCR and Control Building EBR chillers when in standby to ensure the chillers will automatically start and remain in service to remove Control Building heat loads with ERCW supply temperature below 65 degrees F. Any potential failure modes introduced by the installation of the metering valves will cause the TCVs to fail to positively isolate ERCW flow. Monitoring the pressure between the two metering valves should ensure that these valves have not failed to perform their pressure control function. The modulation of these TCVs during chiller operation is not affected by this alteration. Thus, the valve function during design basis events is not affected. The FSAR failure modes and effects analysis considers one train ERCW TCV failure closed to be a credible failure with no adverse impact. This change does not increase the probability of this failure as the modulation function of this actuator is not affected by this change. No common mode failures are introduced by this change. This change has been recommended by the valve manufacturer, METREX.

Verify TCV associated with the standby chiller has near zero leakage across the valve seat.
Verify chiller will start and operate.
During operation chiller suction and discharge pressures will be monitored to ensure they remain stable and acceptable.

Listed below are the affected TCVs and their associated chiller:
0-TCV-067-1052-B, Electric Board Room Chiller B-B
0-TCV-067-1051-A, Main Control Room Chiller A-A
0-TCV-067-1050-A, Electric Board Room Chiller A-A
0-TCV-067-1053-A, Main Control Room Chiller B-B

Design Change Notice (DCN) W-39476-A reviews installation of valves installed per TACF 0-97-003-067. The subject tubing connects to and is part of the pressure boundary for safety related ERCW piping serving the Main Control Room Chillers and the Electrical Board Room Chillers. This ERCW piping is TVA Class C, ASME Class 3. The tubing qualification was performed to ensure that all applicable requirements from design criteria WB-DC-40-31.7 and WB-DC-40-31.9 were met, including material stress allowables and pipe support design. Loading conditions considered included deadweight, thermal, seismic inertia, and seismic anchor movement. Existing stress analysis calculations for the ERCW lines were revised to include documentation of the tubing qualification. Calculations revised were N3-67-16A, N3-67-17A, N3-67-53A, AND N3-67-57A. DCAs against the stress analysis isometric drawings have also been provided to document the configuration analyzed. In addition, the pressure indicators, which were temporary, are installed as permanent plant equipment and supported as required.

SA-SE Number *WBOTSS-97-058-0*

Affected Documents	*Document Type*	*Safety Assessment Title*	*Implementation Date:*
Procedure 1-PI-OPS-1-TB Temporary Procedure TP-06-027	Temporary Procedure	#1 Feedwater Heater Operating Vent Valve Isolation	8/28/97

Description of Change, Test, or Experiments

This temporary procedure TP-06-027, #1 Feedwater Heater (FWH) operating vent valve isolation, provides the steps necessary to close and reopen the #1 FWH's operating vent valves in order to perform maintenance activities on downstream portions of piping. The procedure also includes appropriate monitoring activities to ensure feedwater temperature is not adversely affected during the closure period. This procedure may be used to determine the duration these valves can remain closed during full power operation without significantly affecting heater performance. It may also be used to determine the length of time the subject vent valves should be opened following a closure period to effectively vent any collected non-condensables to the condenser. The SAR Section 10.4.10.2 describes the subject vent valve alignment during normal operation. The SAR states in part, "venting [of the feedwater heaters] to the main condenser during normal operation is accomplished through continuous 'free blowing' orifices...". This temporary procedure will close the subject vent line for a period of time during normal operation; thereby, conflicting with the reference. Section I5.2.10 of the SAR addresses excess heat removal events caused by feedwater system malfunctions. Although not specifically addressed in the SAR, WB-DC-40-64, Section 4.23.1 defines the event as "secondary system malfunctions which cause excessive feedwater addition or reduction in feedwater temperature. The resultant effort is an unplanned increase in heat removal by the secondary system, accompanied by a decrease in reactor coolant temperature which increases core reactivity and can lead to an increase in core power. Either a control system malfunction or operator error may be the initiating event." In addition, SER Section 15.2.2 lists "decrease in feedwater temperature" as an event that WBN has analyzed which produced increased primary system cooling. Although WB-DC-40-64 addresses the defined event as caused by excessive feedwater addition in detail, it does not address the defined event as caused by a reduction in feedwater temperature. The Accident Analysis Parameter Checklist (AAPC) offers insight as to why the defined event as caused by a reduction in feedwater temperature was not addressed. Section 4.1.9.1 of the AAPC states in part that the defined event could result from the "failure of a bypass valve that diverts flow around the low pressure feedwater heaters for plants so equipped. Watts Bar Units 1 and 2, however, are not equipped with this type of bypass valve." The referenced section of the SAR is pertinent to the subject temporary procedure since the closing the subject vent line for an extended period of time could lead to air binding of the #1 heaters and a subsequent reduction of heat transfer to the feedwater system. If allowed to progress, final feedwater temperature could be reduced. The subject procedure administratively limits the decrease in the #1 heater's outlet temperature to 3 °F and the decrease in final feedwater temperature to 1.5°F. Although not explicitly bounded by the SAR or by the Design Criteria, the 1.5°F decrease limit has been determined to not impact the analysis described in the SAR. Specifically, when questioned about the subject activity, Westinghouse stated "Since the overall feedwater temperature should change by no more than 1.5°F and the temperature reduction will be periodic, it is believed that the temperature reduction will have an negligible impact on the licensing basis LBLOCA, SBLOCA, SGTR and Transient Analyses. Furthermore, there should be no impact on the containment integrity analysis nor in the steamline break mass and energy releases. Also, there should be no impact on the RTS/ESFAS setpoints or initial condition uncertainties."

Safety Evaluation Summary

This temporary procedure does not involve an unreviewed safety question. The controlled closure of these valves will not create the possibility of any other type accident or any additional failure modes. The only credible type of accident that closing these vent valves could be associated with is an Excessive Heat Removal Due To Feedwater System Malfunctions accident. Since the overall feedwater temperature change is administratively limited to no more than 1.5 F reduction and the temperature reduction will be periodic, the temperature reduction will have an negligible impact on the licensing basis accidents evaluated previously in the SAR.

The controlled closure of these valves does not increase the probability of occurrence of a malfunction of equipment important to safety. These are non-quality related valves between non-quality related components. All required equipment important to safety will remain in-service during this activity.

The controlled closure of these valves will not increase the consequences of an accident previously evaluated. The closure of these valves does not change the reactor coolant system's accident mitigation performance and therefore does not increase the potential for radiation releases to the environment.

Based on these arguments, the change does not involve an unreviewed safety question.

SA-SE Number *WBOTSS-97-077-0*

Affected Documents	*Document Type*	*Safety Assessment Title*	*Implementation Date:*
WO 97-00-7350-000 TS Bases 97-003 Bases 3.3.2 & B3.3.6 & Bases Rev. 9	Work Order	Missed Surveillance Due to Inadequate Surveillance Instruction, 1-SI-99-5	5/5/97

Description of Change, Test, or Experiments

Due to the inadequate surveillance instruction, 1-SI-99-5, SRs 3.3.2.8 and 3.3.6.6 were not completely tested and thus this problem constitutes a missed surveillance. WO 97-007350-000 will perform testing to satisfy missed SR 3.3.2.8, TRIP ACTUATING DEVICE OPERATIONAL TEST (TADOT), with a frequency of 18 months for LCO 3.3.2 Function 2.a Containment Spray (CS) Manual Initiation and Function 3.a.(1) Containment Isolation Phase A (CIA) Manual Initiation, and SR 3.3.6.6, TADOT, with a frequency of 18 months for LCO 3.3.6 for Function 1 Containment Vent Isolation (CVI) Manual Initiation. These functions were completely tested by Preoperational Test Instruction 99-04 in October 1994.

Prior to testing the CS Manual Initiation function each switch will be verified open for CS, CIB, and CVI by checking for zero volts across each contact at the Main Control Board (MCB) field side terminal blocks prior to actuating each switch to prevent an inadvertent actuation of CS, CIB, and CVI. Then the first switch will be actuated and its CS contacts verified closed. When the switch is released to normal its contacts will again be verified open prior to proceeding to the next switch. When all four CS switches are tested train A SSPS will be removed from service and the manual initiation CIA and CVI input wires will be lifted. Train A SSPS will be restored to service and then train B SSPS removed from service and the manual initiation CIA and CVI input wires will be lifted. Each of the CIA & CVI manual initiation switches will be tested for contact closure to the lifted wires at each train of SSPS. When the CIA & CVI switches are tested the wires lifted in train B SSPS will be landed and the master relay to switch contacts at the MCR terminal board continuity verification for CS, CIB CIA and CVI will be completed. Train B SSPS will then be restored to service and train A SSPS removed from service. The wires lifted in train A SSPS will be landed and the master relay to switch contacts at the MCR terminal board continuity verification for CS, CIB, CIA and CVI will be completed. Train A SSPS will then be restored to service and testing completed. Prior to lifting any wires compensatory actions will be put into place to allow the manual initiation of CIA and CVI from the SSPS when directed by the MCR crew. Manual initiation of CIA and CVI from the SSPS slave relays, as proposed by Technical Specification Bases change 97-003, is another method of accomplishing the manual initiation in addition to the switches in the MCR as described in the Tech Spec Bases. The proposed Tech Spec Bases change does not add any addition equipment to the plant or change or add test methods not previously analyzed in the SAR. A thorough crew briefing and pretest briefing will be conducted. An additional operator in the MCR will be dedicated to respond to an inadvertent ESF actuation. There are numerous safeguards built into the conduct of the test. There is no impact on the SAR as the manual initiation functions are not credited in any accident analyzed in the SAR. There are no specific details on testing the manual initiation functions described in the SAR.

Safety Evaluation Summary

Testing as described will be conducted in accordance with the Technical Specification LCOs and as described in the Tech Spec Bases. Proposed Technical Specification Bases change 97-003 provides another method of accomplishing the manual initiation in additional to the switches in the MCR as described in the Tech Spec Bases. The proposed Tech Spec Bases change does not add any addition equipment to the plant or change or add test methods not previously analyzed in the SAR. The use of compensatory measures provides an additional method of accomplishing manual actuations if required. This test will not reduce any margins of safety as defined in the Tech Spec Bases. These manual initiation functions are not credited in any accident event or transient analyzed in the SAR. The likelihood of an event occurring during the short period of time compensatory measures are in effect is extremely small. The further likelihood that the automatic actuations would not occur if required is extremely small. The use of two operators in the AIR to actuate each train of SSPS, and thus each train of safety function, maintains equipment diversity and redundancy. There are numerous safeguards against inadvertent actuations and reactor trip built into the conduct of the test. Lifting and landing of wires will only take place with the associated train out of service, thus precluding an inadvertent ESF actuation or reactor trip. All reasonable efforts have been made to reduce the burden on the operating crew and test performers. As these manual initiation functions are not credited in the SAR accident analysis, Primary and Secondary Containment Integrity/Isolation will be maintained as described in the SAR. The Safety Injection functions are not impacted by this test. All automatic ESF functions will remain available in accordance with Tech Specs throughout this test. Manual initiation of CIA and CVI from the SSPS slave relays is another method of accomplishing the manual initiation in additional to the switches in the MCR as described in the Tech Spec Bases.

SA-SE Number *WBOTSS-97-115-0*

Affected Documents	*Document Type*	*Safety Assessment Title*	*Implementation Date:*
TACF 0-97-05-030	Temporary Alteration	Temporary Alteration to Permit Fire Dampers to Close and HVAC Intake Opening Around Spent Fuel Pit to be Covered	7/21/97

Description of Change, Test, or Experiments

TACF 0-97-05-030 permits modification of the AB HVAC system. In order to alter normal air flows during modification of Spent Fuel Racks, the following HVAC changes are necessary to eliminate the potential spread of contamination into the ventilation ductwork.
Fire dampers 0-ISD-031-3847, 0ISD-031-3848 would be closed.
All 72 of the HVAC openings around the Spent Fuel Pit at elevation 750' ½" will be blocked closed using herculite and duct tape.
Equipment Hatch located on refuel floor, column A8 & V, would be opened to permit refuel floor and el 737 area to be at equal pressure.

Activity in SFP is associated with implementation of DCN M 38623 to modify the Spent Fuel Racks. During this activity the SFP will not be filled with water. There is currently no radioactive material stored in the SFP. The Spent Fuel racks which are to be installed per this DCN have been removed from SQN. These racks have some fixed contamination which may be freed during the installation process. Upon completion of rack installation these temporary changes will be removed and this TACF will be closed. This will be done prior to declaring the spent fuel pool cooling system operational.

The intent of this TACF is to make temporary modifications to the ventilation system to ensure that if any contaminated material is freed during work on the Spent Fuel racks it will not be drawn into the permanent HVAC system. These changes and the temporary conditions implemented by the work documents associated with DCN implementation will contain any potentially contaminated material which may be freed during installation of Spent Fuel racks.

During routine plant operation one of the design functions of the Fuel Handling Exhaust system is to maintain the refuel floor at a negative pressure. The specified negative pressure is maintained using modulating dampers associated with the FHE system. The lower elevations of the AB are maintained at the same specified negative pressure by air exhausted using the AB General Exhaust system. The specified pressure is maintained using modulating dampers associated with the AB General Exhaust system.

Opening the hatch cover will permit the modulating dampers associated the AB General Exhaust system to aide in maintaining the required pressure on the refuel floor.

During plant operation, if an Auxiliary Building Isolation (ABI) signal is initiated the design function of the Fuel Handling Exhaust system and the AB General Ventilation system is to stop. Isolation of the AB during an ABI is provided by dampers not impacted by this TACF.

During an ABI the AB pressure is maintained using the Auxiliary Building Gas Treatment (ABGT) system.

Some duct which is a part of the FHE system is shared by the ABGT system. As described on the attached Table A, the blockage of air flow from around the SFP does not have a significant impact to

Safety Evaluation Summary

This change permits continued operation of the FHE system during work in the SFP, reference DCN M-38623.
The FHE system is not safety related, the changes to this system will not adversely impact plant safe shutdown, nor will they impact plant operability. System will continue to perform it's design function while temporary change is in place. System components will not be adversely impacted by the temporary changes.

The ABGTS is the only Tech Spec system affected by this TACF. Covering of the air exhaust openings will have no measurable affects on the ability of ABGTS to perform its safety related function with respect to maintaining off site dose within limits in the event of an accident prior to the restoration of the TACF. Table A provides details of ABGT system impacts. The margin of safety for this system is not degraded by these changes.

The cover over the equipment hatch in the el 757 floor serves no safety related function. It has no function during a HELB (High Energy Line Break) or MELB (Moderate Energy Line Break) scenario.

The changes made by this TACF are temporary and will be removed prior to Unit 1 refueling outage.

This change does not increase the probability or consequences of any failure of safety related components.

the operation of the ABGT system.

The SAR impacts of the proposed changes are temporary. These changes will remain in place for the duration of activities related to Spent Fuel rack installation. These changes will be removed prior to cycle 1 refueling outage. Specific SAR impacts are listed below:

a) The FHE system total air flow rate is slightly decreased. The air flow rate in the unmodified portions of the FHE system will increase. System modulating dampers will open to exhaust air flow to maintain refuel floor at the required negative pressure. Also, the removal of the equipment hatch cover will permit AB GE system fans to aid in maintaining the refuel floor at the required negative pressure.
b) SAR figures 6.2.3-16 and 9.4-8 refer to TVA DWG NO 1-47W866-10. This drawing identifies the configuration and flow rates of the FHE system and the ABGT system. Fire dampers ~ISD-031-3847 and 0-ISD-031-3848 will be closed. All HVAC openings around the Spent Fuel Pit will be blocked, air flow rates in the 2 ducts from the SFP are reduced to 0, and air flow rates from the remaining FHE ducts increase slightly.
c) The designed "air curtain" across the SFP will not occur. This feature of the SAR is not a required function at this time because there will be no water in the SFP and no fuel in the SFP while this TACF is in effect.

The design function of the "air curtain" is to ensure any potentially contaminated particles originating in the Spent Fuel pit are drawn directly into the ventilation system. This TACF and temporary changes (cover over SFP and installed temporary fans with HEPA filter) implemented by work documents associated with DCN M-38623 insures that any potentially contaminated particles are captured.

SA-SE Number *WBOTSS-97-141-0*

Affected Documents	*Document Type*	*Safety Assessment Title*	*Implementation Date:*
DCN S-39238-A FSAR Change Pkg. 1481	FSAR	Minor Discrepancies Between the FSAR and the Operating Configuration of the Ice Condenser System	9/7/97

Description of Change, Test, or Experiments

The change being considered is FSAR Change Package #1481 and revisions to System Description N3-614001 for consistency. This change package makes a number of minor changes throughout the FSAR in response to PER WBPER970113 which documented discrepancies in the FSAR and the operating methodology of the Ice Condenser System. The following is a list of the changes being made for which a safety evaluation is necessary:

1. Addition of accident function for the floor drains in the Ice Condenser in Section 6.7.1.1 (Floor Drain). Addition of this function clarifies the accident requirements for the system.

2. Deleted reference to the maximum expected pressure in the floor cooling loop of the Ice Condenser in Section 6.7.1.2 (Wear Slab and Floor Cooling System) and 6.7.1.3 (Floor Cooling System). The value previously included in the FSAR was in error and including this pressure in the FSAR does not provide any additional guidance for correct operation of the system.

3. Corrected the average ice bed temperature range from 10 to 1 5°F to 15 to 20°F. This change more accurately reflects the actual operating temperatures seen in the ice bed which typically range in the 18 to 19°F. This change prevents the misunderstanding of operating the ice bed at too warm of a temperature

4. Corrected upper plenum temperature in the Ice Condenser from 15° to be between 15° and 20°F in Section 6.7.6.2 (Third Stage - Air Cooling Loop). This change prevents the possibility of misunderstanding the operating temperature of the ice bed.

5. Deleted reference to temperature modulating valve on the Air Handling Units (AHUs) in the upper plenum of the Ice Condenser from Section 6.7.6.2 (Third Stage - Air Cooling Loop). This valve has never existed at Watts Bar and this information was left in the FSAR in error.

6. Deleted reference to the length of the defrost cycle for the AHUs in the upper plenum of the Ice Condenser from Section from Section 6.7.6.2 (Third Stage - Air Cooling Loop). This value should not be specified in the FSAR since a continuous process is underway to determine the most appropriate duration to service the needs of the Ice Condenser.

7. Deleted the discussion on defrosting the entire Ice Condenser from Section 6.7.15.3 (Design Evaluation). Watts Bar does not intend to perform a complete Ice Condenser defrost due to the possibility of water accumulation in the floor and the potential of freezing baskets in the lattice frames. Watts Bar is currently examining other methods of frost removal from the wall panels to eliminate the need of a defrost cycle.

8. Revised the maximum design glycol pressure from 150 to 180 psig in Table 6.7-18. This change will encompass the actual pressures observed in the field.

9. Revised the temperature of the ice bed and static air from1 5°F to a range of 18° to 1 9°F,

Safety Evaluation Summary

During normal plant operations, the Ice Condenser System is required to be operable, but has no active safety function; it is a passive system which must operate only during an emergency. The safety function of the system is to absorb thermal energy released during a LOCA or HELB. The changes summarized in Section A. 1 do not have the potential to impact safe operation of the Ice Condenser System. All changes described are minor clarifications to prevent the potential for misunderstanding and to bring the FSAR into agreement with actual field conditions/configurations which have been determined over time at both SQN and WBN to be the best suited to maintain the Ice Condenser in the optimum condition available.

nominal. These are the normally observed temperatures in the ice bed during operation and this should preclude the possibility of a misunderstanding of the correct ice/air temperature.

10. Added the UNIDs for the temperature elements, corrected RTD elevations, and corrected detail reference numbers in Table 6.7-24.

11. Corrected the RTD locations shown in Figure 6.7-38 to agree with the actual installed field locations.

12. Updated TVA drawing shown in Figure 6.741 to latest revision.

13. Permits/inch unions to be added to the glycol cooled floor piping per DCN S-39328-A. This permits valves 1-SPV-61-696 through -742 (even numbers only) to be more easily removed and replaced on a maintenance as-needed basis.

SA-SE Number *WBOTSS-97-194-0*

Affected Documents	*Document Type*	*Safety Assessment Title*	*Implementation Date:*
TACF No. 1-97-19-6 WR/WO 97-010976-000 (Complete 10-16-97)	Temporary Alteration	Temporary Alteration to Disable 1-RTV-6-1538A in the Closed Position	10/17/97

Description of Change, Test, or Experiments

This temporary alteration (TACF) will disable 1-RTV-6-1538A in the closed position in accordance with WO 97-0109760-000 by "killing" the valve. This is done by injecting Furmanite compound upstream of the leak per Furmanite procedure N-97220. It is important to note that this valve is in an essentially dead leg serving the referenced float type level switch; i.e., no flow condition. The valve will be restored to it's normal configuration / replaced no later than RF01.

This temporary alteration will also configure the lower root valve to the subject level switch, 1-RTV-6-1539A, in the closed position in order to isolate the level switch until the permanent repair / replacement is performed.

The field wires at 1-LS-6-28B will be lifted, lugged, and taped to prevent common abnormal level alarm window XA-55-2B-30D from erroneously alarming due to the subject level switch removal-from-service and to allow the high level alarm to remain functional.

The SAR, section 10.4.10.2, describes the subject MSR drain tank level alarm function: "Low level alarm is also annunciated if the level drops below the normal control range." This TACF defeats the subject alarm for a period of time during normal operation; thereby, conflicting with the reference.

The only type of accident that defeating this alarm could be associated with is "Minor Secondary System Pipe Breaks" . The potential effects of defeating this level switch are considered insignificant. Since the normal LCV fails closed on a loss of air to the actuator or level controller, this would cause the bypass LCV to open to control level and potentially the high level alarm to actuate. The impact of this failure mode is a reduction in plant efficiency. If the normal LCV failed open due to a malfunction, steam flow would increase through the LP tube bundles and a steam and liquid mixture would pass through the drain line to the number 2 condensate heaters. The adverse impact is principally a reduction in plant efficiency due to higher extraction steam flow. A slightly higher potential for flow accelerated corrosion may exist in the drain line. The increased flow condition would be detected by monitoring the level gauge or by system thermal performance monitoring within a shift to a few days. The adverse impact of the duration of this condition on the piping is considered insignificant. Even if the potential increased erosion in the 6 inch MSR LP drain line was considered credible, a complete loss of this line is bounded by current analysis. The SER does not describe this system in any manner in chapter 10 nor does it address minor secondary system pipe breaks in chapter 15.

Safety Evaluation Summary

This temporary alteration does not involve an unreviewed safety question. The defeat of this level switch will not create the possibility of any other type accident or any additional failure modes. The only credible detrimental affect of defeating this level switch would be a slightly higher potential for flow accelerated corrosion in the associated drain line. The potential affects of defeating this level switch are discussed in the following paragraph.

Since the normal LCV fails closed on a loss of air to the actuator or level controller, this would cause the bypass LCV to open to control level and potentially the high level alarm to actuate. The impact of this failure mode is a reduction in plant efficiency. If the normal LCV failed open due to a malfunction, steam flow would increase through the LP tube bundles and a steam and liquid mixture would pass through the drain line to the number 2 condensate heaters. The adverse impact is principally a reduction in plant efficiency due to higher extraction steam flow. A slightly higher potential for flow accelerated corrosion may exist in the drain line. The increased flow condition would be detected by monitoring the level gauge or by system thermal performance monitoring within a shift to a few days. The adverse impact of the duration of this condition on the piping is considered insignificant.

The only type of accident that defeating this alarm could be associated with is "Minor Secondary System Pipe Breaks" (Chapter 15.3.2); however, as discussed above, the potential increased erosion of the drain pipe is considered insignificant and will have no impact on the licensing basis accidents evaluated previously in the SAR.

The defeat of this alarm does not increase the probability of occurrence of a malfunction of equipment important to safety. This is a non-quality related level switch for a non-quality related MSR drain tank. All required equipment important to safety will remain in-service during this temporary alteration.

The defeat of this alarm will not increase the consequences of an accident previously evaluated. The defeat of this alarm does not change the reactor coolant system's accident mitigation performance and therefore does not increase the potential for radiation releases to the environment.

Based on these arguments, the change does not involve an unreviewed safety question.

SA-SE Number *WBOTSS-97-241-0*

Affected Documents	*Document Type*	*Safety Assessment Title*	*Implementation Date:*
MI-88.003 Rev. 1	Procedure	Sludge Lancing During Core Alterations and the Installation of Hoses and Cable Associated with Steam Generator Primary Side Maintenance	9/7/97

Description of Change, Test, or Experiments

The subject procedure revision identifies requirements for sludge lancing during core alterations and the installation of hoses and cables associated with Steam Generator primary side maintenance. This includes temporarily installing hoses and cables in a containment penetration during an outage (unit in mode 5, 6 or core empty). This Safety Evaluation demonstrates that these penetrations can be in an altered configuration and that selected temporary fluid and gas filled hoses can remain in service during core alterations and provide a barrier between containment and the outside atmosphere in the event of a fuel handling accident or loss of RHR shutdown cooling precluding an unfiltered release of radioactive material to the public.

Sludge lancing uses an equipment system that temporarily locates equipment both inside and outside containment and requires a containment penetration for connection hoses and cables. Steam Generator Eddy Current examinations also require equipment located both inside and outside containment with associated connecting cables. Also compressed air hoses are required to power the sludge lance return pumps and is also used to cool the eddy current equipment. The sludge lance connecting hoses contain 1) high pressure cleaning water entering containment, 2) cleaning water (with entrained sludge and air) exiting containment, and 3) air (from the surge tank and holding tank) vented back into containment. The sludge lance cables are for equipment control wiring and communications. The eddy current cables are for platform cameras. communications, and to transmit the eddy current data out of containment.

While in mode 5 these hoses and cables are installed in a penetration and the interstitial spaces filled with approximately 19 inches of RTV foam. Manual isolation valves are installed near the penetration on the hoses inside containment and the annulus as a minimum.

Revision 1 of MI-88.003 incorporates the following changes:

a) Revised to allow the hoses, power, control and signal cables temporarily installed in Containment and Shield Building penetrations X-54, X-117, X- 118, MK-72, MK-99 during mode 5, 6 or core empty to remain temporarily installed during core alterations, handling of irradiated fuel inside containment or the fuel handling area, and mid-loop operations.
b) Revised to allow the manual isolation valves on fluid and gas filled hoses used for S/G sludge lancing at either side of the Containment and Shield Building penetrations X-54, X-117, X-118, MK-72, MK-99 during mode 5, 6 or core empty to remain open during core alterations, handling of irradiated fuel inside containment or the fuel handling area, and mid-loop operations provided a sufficient barrier has been verified intact between containment atmosphere and the outside environment. Prompt closure of these manual isolation valves is required when this barrier integrity is lost, prior to breaching the barrier established with the sludge lance equipment, in the event of a loss of RHR shutdown cooling during mid-loop operations or a fuel handling accident inside containment or the fuel handling area.
c) Revised to provide additional emergency closure actions to be performed should a loss of RHR cooling occur when Containment penetrations are not intact during mid-loop operations. These steps

Safety Evaluation Summary

The WBN Unit 1 Technical Specification Bases states:

"In MODE 6, the potential for containment pressurization as a result of an accident is not likely; therefore, requirements to isolate the containment from the outside atmosphere can be less stringent. The LCO requirements are referred to as 'containment closure' rather than 'containment OPERABILITY.' Containment closure means that all potential escape paths are closed or capable of being closed. Since there is no potential for containment pressurization, the Appendix J leakage criteria and tests are not required....

The requirements for containment penetration closure ensure that a release of fission product radioactivity within containment will be restricted from escaping to the environment. The closure restrictions are sufficient to restrict fission product radioactivity release from containment due to a fuel handling accident during refueling....

The other containment penetrations that provide direct access from containment atmosphere to outside atmosphere must be isolated on at least one side. Isolation may be achieved by an OPERABLE automatic isolation valve, or by a manual isolation valve, blind flange, or equivalent. Equivalent isolation methods must be NRC approved and may include use of a material that can provide a temporary, atmosphere pressure, ventilation barrier for the other containment penetrations during fuel movements....

The affected penetrations shall only be altered in modes 5 and 6 when containment integrity and Shield Building integrity are not required. No risk of internal flood, moderate or high energy line break exists in this condition. No safety related cables penetrate the affected penetrations. These altered penetrations have no impact on Appendix R safe shutdown equipment in modes 5 and 6. The interstitial spaces between the hoses and cables will be filled with at least 12 inches of RTV foam per design drawings and consistent with Tech Spec 3.9.4 bases. When penetrations are breached for installation and removal of cables and hoses, breaches to ABSCE and Containment Closure are tracked per TI-65 and TI-68.003 to ensure these breaches are closed prior to performing Core Alterations or movement of irradiated fuel inside containment of the fuel handling area. This instruction is considered an emergency closure procedure per GL 88-17 per AOI-14, Loss of RHR Shutdown Cooling and TI-68.002 when penetrations are breached during mid-loop operations. Configuration control shall be maintained over sludge lancing equipment when the altered penetrations are sealed to ensure no path exists from containment to outside atmosphere. Closure requirements are consistent with GL 88-17 definition. Upon notification of a Fuel Handling Accident or loss of RHR Shutdown cooling during mid-loop operations both manual isolation valves on each hose shall be closed. Within 4 hours following a fuel handling accident, the affected penetrations will be restored using one permanent flange to assure no unfiltered radioactive release path is created due to a seismic event. Operability for containment integrity is established prior to mode 4 entry by performance of LLRT's on testable penetrations. Contingencies have been developed to restore altered penetrations to normal upon notification of require removal or cutting of hoses, power and signal cables in the penetrations and the reinstallation of at least one blind flange in the annulus. These flanges arc testable and equipped with two seal rings. All tools required to perform these actions shall be staged at penetrations X-54, X-117, and X-118. This emergency closure procedure is tracked by TI-68.002 when penetrations X-54, X-117, and X-118 are breached.

d) Revised to provide actions to be performed should a fuel handling accident or loss of RHR shutdown cooling in mid-loop operation occur, to ensure design basis closure provisions which conservatively preclude an unfiltered release of radioactive material for other occurrences within the next 100 days, such as an earthquake, flood or tornado.

e) Revised to direct restoration of the penetrations below flood elevation (X-118, MK-99) to "normal" upon notification of a flood. These penetrations are more than 7 feet below the design basis flood elevation of 740.1 feet and would otherwise not prevent flooding into the annulus or containment when altered by this instruction. The remaining penetrations will withstand the DBE flood in their temporary configuration, but closure should be restored to normal as directed by the Shift Manager. All penetrations shall be restored to their normal configuration prior to mode 4 entry or power ascension. These penetrations shall be restored within27 hours consistent with the notification period for a design basis flood.

f) Penetrations shall be restored to their normal configuration or the configuration required to support performance of a Containment Integrated Leak Rate Test (CILRT). Local Leak Rate Testing (LLRT) shall be performed on penetration X-117 prior to performance of CILRT.

g) As left LLRT shall be performed on penetrations X-54, X-117, X-118 prior to mode 4 entry or power ascension.

h) Revised to obtain Shift Manager direction for installation of Shield Building penetration MK-72 outside cover upon notification of a tornado watch/warning to eliminate any potential external missile hazard adverse impact on the ABSCE integrity and Containment closure, if required.

i) Revised to direct installation of at least one permanent blind flange on penetrations X-54. X-117, X-118 within 4 hours following a fuel handling accident inside containment and MK-72 within 4 hours following a fuel handling accident in the refueling area.

j) Dow Corning 3-6548 Silicone RTV Foam shall be used to seal all the affected penetrations. A minimum depth of 12 inches is required in all penetrations regardless of cables or hoses. A continuous nominal 1/4 inch bead of Dow Corning adhesive sealant 732, 96-081, 790, or 795 shall be injected at the foam to hose/sleeve interface on one side in X-54 and MK-72. Use of this sealant and adhesive for these penetrations is documented in the following drawings:

X-54	47A472-11 Detail XI (hoses)	45W883-1 Detail HI (cables)
X-117	45W883- 1 Detail H1 (cables)	
X-118	45 W883- 1 Detail H1 (cables)	
MK-72	47A472-11 Detail XI (hoses)	45W883-1 Detail HI (cables)
MK-99	47W600-109 (foam)	45W883-1 Detail HI (cables)
	47W470 Detail N3 (foam)	

These barriers are rated for 3 psi air or flood and 437 degrees F using this sealant. No restrictions apply to the number of hoses or the separation of the hoses or cables in these applications. Maximum cable loading is 50% of the penetration area. As no Kaowool Boards will be installed at these penetrations to ensure a 3 hour fire rating per drawing 45W883-1 Detail 1, installation of these temporary features in the penetrations is considered a fire impairment that must be tracked and restored prior to mode 4 entry. All penetration seals shall be smoke tested to ensure that barrier integrity has been established following alteration and prior to declaring the breach restored for TI-65 and TI-68.002.

k) Manual isolation valves shall be installed on all hoses which penetrate X-54 and MK-72 inside containment, inside the annulus, and outside the Shield Building. Configuration control of these valves shall be maintained in this instruction. During fuel handling or core alterations, personnel shall other design basis events such as flooding and tornadoes which could adversely impact the altered penetration integrity.

be assigned to close manual isolation valves on hoses inside containment and the annulus which penetrate X-54, MK-72 to ensure immediate close upon notification of the event.

l) The worst case scenario for containment pressurization and heatup is the loss of RHR shutdown cooling. RCS pressurization is limited to 2 psig by the RCS vent paths available during mid-loop operations due to minimal decay and residual heat. Thus, containment pressure should never achieve 2 psig or saturation temperature for 2 psig water due to core boiling of 229 degrees F per WB-DC-40-64. All penetrations altered by this instruction exceed these pressure and temperature conditions.

m) The temporary hoses and related isolation valves and equipment that establish the barrier between containment and outside atmosphere for sludge lancing S/G's are commercial grade and rated for at least 150 psig and 250 degrees F. These components do not satisfy seismic category I, TVA safety class B requirements. However, these components are not used to maintain containment integrity as required in modes 1 through 4. These components will be disconnected in the event of a flood and normal closures established in the event of a tornado warning (MK-72). Normal seismic category 1, TVA safety class B barriers will be restored within 4 hours following a fuel handling accident or loss of RHR shutdown cooling during mid-loop operations. No Appendix R fires or internal missile hazards need to be considered for the use of this equipment as it will only be used during modes 5 and 6 when moderate or high energy line breaks or fires affecting safe shutdown equipment are not postulated to occur. An in service leak; test of the sludge lancing equipment ant related hoses shall be performed following alteration of X-54 and MK-72 and prior to declaring the breaches restored per TI-65 and TI-68.002.

n) During installation and restoration of the temporary alterations to these penetrations, breaches shall be tracked by TI-65 for ABGTS operability per Tech Spec 3.7.12 and TI-68.002 for Containment penetrations operability per Tech Spec 3.9.4 and NRC GL 88- 17.

o) Acceptable power, control and signal cables for use in the affected penetrations are defined by WB-DC-30-5, Table 9-3 and Table 9-3A, which lists acceptable cables for use inside primary containment. These cables are also bound by analysis per drawing 45W883-1 for use in the penetrations. Load rating of power cables used in these penetrations shall exceed the maximum anticipated load per DS-E12.6.3.

SA-SE Number *WBPLCE-96-048-3*

Affected Documents	*Document Type*	*Safety Assessment Title*	*Implementation Date:*
DCN M-38623-A FSAR Change Pkg. 1464	DCN	Spent Fuel Storage	9/8/97

Description of Change, Test, or Experiments

DCN M-38623-A makes changes to the Fuel Handling and Storage System which provide a total spent fuel assembly storage of 1835 storage locations within the spent fuel pool. The change includes:

1. Removal of the existing Wachter Associates designed spent fuel storage racks (SFSRs) and immediate replacement with SFSRs designed by Programmed and Remote (PaR) Systems Corporation and future installation of additional new SFSRs designed by Holtec International. The PaR racks were formerly used at Sequoyah and were replaced with new racks to provide increased storage capacity. All analyses, supporting documents, and licensing submittals are based on a total of 1835 storage locations, which includes the future installation of the additional racks. The PaR racks are modified by venting the neutron absorber cavity to allow water to enter the cavity.

After installation in the pool, the PaR racks will be drag tested using a dummy fuel assembly and a calibrated load cell. Cells exceeding a 50 pound load test criteria will be reworked with a specially designed expander tool and retested to confirm cell envelope acceptability before release as an approved spent fuel assembly storage location. In order to support receipt of new fuel, cell drag testing and storage system return to service may be done in stages. Stage 1 would be to complete physical work as outlined in DCN M-38623-A except index strips and tool brackets and drag test a minimum of 24 cells. Stage 2 would be to drag test a minimum of 358 cells. Stage 3 would be to complete the index strips and tool brackets and drag test the remaining cells which are accessible without removing the wall obstructions. Any cells which are not drag tested under DCN M-38623-A shall be documented in Surveillance Instruction 0-SI-79-1, "Verification of Fuel Storage Configurations". It is anticipated that 10 to 20 cells which are shadowed by the strainers, skimmers and a temperature sensor/indicator may not be accessible for drag testing. These obstructions have been flanged and can be removed. Therefore, when the obstructions are removed, the cells will be drag tested prior to spent fuel being stored in them.

2. Removal of approximately 64 feet of the sparger/diffuser pipe from the spent fuel pool to allow additional space for the new fuel storage racks. This reduces the SFPCCS's hydraulic resistance very slightly. The pumps will be tested for proper operation and the cooling system will be performance tested under a System Test to ensure adequate flow and bulk temperature control.

3. Fabrication and use of a cask pit shield to provide additional protection against the effects of a load drop accident for crane usage over the cask pit storage area when the new rack contains stored fuel. The lifts and crane movements over the cask pit shield will be controlled by Technical Specification requirements.

4. Implementation of the following supporting modifications related to the use of the new high density storage racks:

a. Future installation of a fuel storage rack support pedestal in the cask pit storage area. This pedestal will support a new rack in the cask pit area at essentially the same elevation as the racks in the spent fuel pool.

Safety Evaluation Summary

The evaluation of postulated accidents with respect to nuclear criticality and/or radioactivity release has shown acceptable results in that keff does not exceed 0.95, including uncertainties, and that postulated releases do not exceed existing regulatory limits. TVA has determined that the proposed storage expansion:

a. Meets the NRC mandated subcriticality at the temperature of stored fuel assemblies.

b. Maintains the peak pool bulk temperature below the threshold value to preclude local boiling.

c. Does not result in rack-to-rack impact in the active fuel cell area or rack-to-wall impact during postulated seismic events.

d. Does not result in increase of reactivity above the NRC guidelines due to postulated accident conditions.

e. Does not overstress the spent fuel pool structure.

f. Does not entail an unresolved safety issue.

The Watts Bar reracking is consistent with other utilities rerack modifications. In addition, the Watts Bar spent fuel pool rerack involves both replacing existing and adding new racks where space permits. The design of the replacement and new spent fuel racks contains a neutron absorber, Boral, to allow close storage of spent fuel assemblies while ensuring that the keff remains less than 0.95 under all operating conditions. Venting the neutron absorber cavity allows water to enter the cavity and ensures that the important water gaps are maintained.

b. Installation of new SFP underwater lighting. The new lighting utilizes high pressure sodium (HPS) technology to provide greater illumination at lower overall power requirements. Five existing lighting locations will be retrofitted with the new lighting assemblies, plus an additional lighting assembly will be added to the SFP/cask pit divider wall. This additional lighting fixture will be designed such that either the SFP or the cask pit can be illuminated by appropriate mounting of the lighting fixture.

c. Addition of new fuel location index strips to the SFP which will correspond to the cell locations of the replacement SFP storage racks. One set of index strips will be attached to the bridge on the fuel handling crane, for North-South location grid locations, and another index strip set will be attached to the SFP curbing running East-West to provide corresponding grid locations for fuel movement operations.

d. Removal of existing fuel and burnable poison handling tool storage brackets and replacement with new brackets similar in design. The new brackets will be added to a moveable support which will span the spent fuel pool/refueling canal divider wall. A fuel handling tool storage bracket will also be added to the SFP handling crane and to the cask pit wall for optional handling tool storage locations.

e. Permanent modification of gate guide brackets by trimming away interfering material . The existing brackets impose on space needed for the new storage racks and for insertion- of the spent fuel assemblies into the storage cell.

f. Addition of flanged connections to the sparger pipe, the strainers and the skimmers. The flanges will allow these items to be removed to provide access to the spent fuel storage cells directly below.

5. Implementation of Technical Specification change WBN-TS-96-010 and several FSAR text, table, graph and figure revisions.

SA-SE Number *WBPLCE-97-013-0*

Affected Documents	*Document Type*	*Safety Assessment Title*	*Implementation Date:*
FSAR Change Pkg. 1475	FSAR	Parking log reclamation and formation of berms over abandoned building slabs	11/6/97

Description of Change, Test, or Experiments

The proposed change consists of minor revisions to FSAR Section 2.4.2, deletion of several FSAR Figures associated with that section, and implementation of Work Order 97-07040-00.

FSAR CHANGE

The FSAR change is intended to remove unnecessary information in the form of figures and make minor revisions to the text of Section 2.4.2 where these figures are referenced. Specifically, Figures 2.4-40a Sheets 2 & 3 and 2.4-40c will be deleted. The only purpose of Figure 2.4-40a Sheets 2 & 3 is to show paved and unpaved areas. Figure 2.4-40c is an expanded view of the larger plant area depicted in Figure 2.4-40b. The additional information shown on the expanded drawing is not germane to the text where that figure is referenced. The figures to be deleted do not contain any information required by R.G. 1.70. All key graphical information pertinent to that section is already shown on other figures which will remain.

The FSAR text in Section 2.4.2 will be revised to reflect the above drawing deletions. The sentence referencing Figures 2.4-40a Sheets 2 & 3 will be deleted. The reference to Figure 2.4-40c will be changed to Figure 2.4-40b.

WORK ORDER 97-07040-00

This Work Order has been prepared to perform reclamation work on old site parking areas and temporary ounce building areas. These areas will no longer be used for those purposes. Affected parking areas currently surfaced with gravel will have the gravel removed and replaced with an equivalent thickness of topsoil. In areas where abandoned buildings have been razed, leaving bare concrete slabs, gravel removed from the parking lots will be placed to create a covering. Topsoil will be added on top of the gravel. These changes are detailed on TVA drawings 185-10-1 and 185-10-2.

All of the proposed reclamation work is outside the security perimeter nuisance fence. All affected areas are specifically listed on the BP-354 Exclusion List and therefore are not under configuration control.

The sketch provided on the next page has been included to assist in describing the proposed changes. The area shown is immediately outside the north perimeter nuisance fence. Compass direction North is toward the top of the page. Areas designated with letters 'A' through 'D' are affected by the proposed changes.

This SA/SE has been prepared to document consideration of the impact of these changes on previous analyses performed on the adequacy of drainage provisions made to accommodate the Probable Maximum Precipitation (PMP).

The drainage provisions are in place to ensure that flooding resulting from the site maximum projected short term rainfall will not prevent safe shutdown of the unit. The referenced calculation has

Safety Evaluation Summary

No accident analyses or equipment are affected by these changes since maximum PMP flood levels will remain unchanged and the consequences and severity of the evaluated natural occurrence will not be increased. These changes will not reduce runoff capability, flow directions or flow quantity, or otherwise affect the results of the PMP flooding evaluation.

No impacts to the Technical Specifications have been identified. Based on these findings, it has been concluded that no reduction in the margin of safety will result, and as such, does not involve an unreviewed safety question.

been performed to show that adequate drainage is in place to prevent unacceptable flooding. Changes to the drainage area topography are not permitted to reduce available drainage capacity below the minimum required. The calculation does not differentiate between grasses and gravel in considering the surface material in place; therefore, the proposed changes from gravel to grass surfacing will not affect the calculation results.

In the parking areas labeled 'C', gravel will be removed and replaced with topsoil. This activity will not result in any significant change in elevation or surface profile for those areas. Therefore, that portion of this task contains no potential impact on PMP flood drainage.

In area 'B', the placement of gravel and topsoil over existing abandoned slabs will result in creation of a 4½ foot high berm outside the north nuisance fence stretching from the east side of the TSOB to the proposed interim parking area due south of the Fab Shop. An area of increased elevation had already existed in this location because of the slabs. Drainage off this berm will be to both sides with runoff flowing along the foot of the berm in the same direction that runoff from the abandoned slabs previously occurred. This runoff will not be re-directed by creation of the berm. No net change in the amount or direction of runoff will result. The drainage to the west side of the main plant structures is bounded on the east by the north-south centerline of the Aux Building. The reclamation drawings referenced above indicate that this will be unaffected. Therefore, total flow into each watershed will be unchanged as a result of creation of the berm.

In area 'D', an interim parking area is to be added for use during outages. This area will be located between the west end of the aforementioned berm and Shop Storage Warehouse "A." Placement of gravel in this area will result in an increase in elevation. This area is treated as a single drainage area. The proposed parking lot is near the upper edge of that area and will not block or obstruct any designated flow path for PMP runoff. The planned elevation change will not increase the quantity of runoff for this or any other drainage path.

Another berm is planned at area 'A', a location previously designated as a potential future storage site for low level radwaste. This area is approximately 850' north of the north perimeter nuisance fence. In this localized area, the elevation will also increase; however, this area was already approximately 3' higher than the surrounding area. The planned additional height will not affect current drainage paths or increase quantity of runoff.

SA-SE Number *WBPLCE-97-028-0*

Affected Documents

TRM Change Pkg. No. 97-010
TRM 3.7.3
TRM Revision 5

Document Type

Technical Requirements Manual and Bases

Safety Assessment Title

Functional Testing of Snubbers

Implementation Date:

8/29/97

Description of Change, Test, or Experiments

This TRM change has been prepared to permit functional surveillance testing of mechanical snubbers during all modes of plant operation. Currently, testing is permitted only in Modes 5 and 6. Testing in Modes 1 through 4 using the administrative controls included in the proposed change is supported by the conclusions of this SE.

Specifically, Bases section B 3.7.3 will be revised to permit functional testing of individual snubbers under very restrictive administrative controls. Those restrictions will be added to the Bases description for Action 3.7.3.3. TRM Section 3.7.3 will also be revised to note that administrative controls are required for testing under these provisions. Both sections will continue to require that the affected piping be declared inoperable if the snubber cannot be returned to service within 72 hours.

This change is driven in part by the realization that some safety related systems, such as RHR, CCS, ERCW and SFP Cooling, are required to be operable during Modes 5 and 6. For systems such as these, the impact of removal of snubbers for testing during any mode of plant operation is no less than in any other mode. Hence, the prior requirement that snubbers on these systems be tested during Modes 5 or 6 offered no relative improvement or offset in the potential impact on overall plant safety. NRC Generic Letter 90-09 states that the purpose of functional testing is to provide 95% confidence of 90-100% snubber operability. This TRM change preserves the ability of the testing program to establish that confidence level while increasing the likelihood of earlier detection of failed units.

Neither the WBN SER and supplements, the WBN FSAR, or the Technical Specifications address performance of functional testing on snubbers. Therefore, the proposed change to TSR 3.7.3 remains consistent with the WBN licensing basis.

Safety Evaluation Summary

The TRM currently provides a 72 hour period for inoperable snubbers to be returned to operable status, either by repair or replacement. Only failure to do so will result in the affected piping being declared inoperable. This approach reflects the relative insensitivity of a seismically supported piping system to the inoperability of a single seismic support. Loads on the piping other than seismic, such as deadweight and thermal, are unaffected since snubbers do nothing to resist these loads.

Stated differently, snubbers in seismically qualified piping systems function solely to increase the resistance of the piping to dynamic accelerations resulting from seismic events or other transients. In permitting the removal of individual snubbers for functional testing during all modes of plant operation, this TRM change reflects the conclusion previously reached in the Bases Safety Analysis, that "Further, various Probabilistic Risk Assessment(PRA) studies have indicated that snubbers are not of prime importance in a risk significant sequence. Therefore, the function of snubbers is not essential in mitigating the consequences of a DBA or transient (Ref. 6)." That conclusion reflected engineering experience with piping subjected to both simulated and natural dynamic excitation. Piping systems have inherent flexibility and toughness which greatly enhance their ability to resist damage from dynamic accelerations beyond that which might be analytically predicted. This is due in part to the conservative approach applied in determining piping response.

Design of seismically qualified piping systems is based on application of overlapping conservatisms, including:

· Use of enveloping seismic spectra developed with conservative combination of closely spaced modes and damping values well below those established by testing. ~ Design margins available in existing qualification calculations.
· Significant additional stress allowable margins provided by operability criteria applicable for use under faulted conditions per ASME Section III Appendix F.
· Design margins available in existing pipe support qualification calculations.

Piping between seismic anchor points such as equipment nozzles or inline anchors is analyzed as a 'seismic subsystem' in the applicable stress analysis calculation. Past evaluations of piping seismic subsystems where an individual pipe support was found to be missing or inoperable have shown that this condition results in only limited impact to the subsystem as a whole. While local stresses might increase beyond those previously calculated, no cases have been identified where the established stress allowable criteria for system operability would have been exceeded. The conclusion reached from these considerations is that the removal of single dynamic supports from piping having multiple other supports will not result in failure of the piping should a dynamic event occur. In the absence of such failure and concomitant loss of pressure boundary, no reduction in nuclear safety will be realized.

The administrative controls introduced as part of this change provide an additional layer of protection by ensuring that any load re-distribution which might result from a dynamic event following removal of a snubber for functional surveillance testing will be isolated from equipment nozzles which might act as seismic anchor points. Other controls provide 'opposite train' protection by limiting snubber removal to a single train. Load increases on inline seismic anchors are limited by controls on multiple snubbers removed simultaneously.

Consideration of the conservatisms inherent in the seismic design of piping and the additional administrative controls noted has resulted in the conclusion that the proposed changes do not reflect any increased potential for failure of the affected piping (failure defined as loss of pressure boundary or position retention capability). Therefore, the proposed change (TR-97-010) to the WBN TRM is acceptable from a nuclear safety standpoint and no unreviewed safety question exists.

SA-SE Number *WBPLCE-98-001-0*

Affected Documents	*Document Type*	*Safety Assessment Title*	*Implementation Date:*
PAI-4.01 (ODCM) Rev. 8	Procedure	Tennessee River Mile Values	1/20/98

Description of Change, Test, or Experiments

During a quality assurance audit at Browns Ferry, auditors identified instances where the location of a single critical feature was identified with at least three (3) different Tennessee River Mile values. As part of the corrective action for a Problem Evaluation Report, the Resource Group conducted a detailed evaluation of locations of critical structures and features along the river. The Resource Group has provided a verified and validated listing of Tennessee River Miles for those locations. Additionally the Resource Group has provided the site licensing organizations with recommended revisions to the FSAR. These changes include: (1) FSAR section 2.4 .12.2.1 - a name change from the Volunteer Army Ammunition Plant to the East Side Utility; (2) FSAR TABLE 2.4-4 - various changes to the Tennessee River Mile for locations and the addition of the Soddy-Daisy Falling Water Utility District, and (3) FSAR TABLE 11.2-10 - the addition of the Soddy-Daisy Falling Water Utility District. In addition, the population doses were updated to reflect the additional water supply, which revises FSAR TABLE 11.2-11. WBN ODCM Table 6.1 is revised to be add the Soddy-Daisy Falling Water Utility District and to be consistent with FSAR Table 11.2-10. Additionally, the ODCM is revised to show that special reports will be submitted to NRC in accordance with 10CFR50.4. These changes are not a result of any physical change to any plant feature or system.

Safety Evaluation Summary

There are not any changes to previously analyzed accident scenarios or any new accident scenarios generated by these changes. The accident and malfunction consequences were not affected. Therefore, there is no unreviewed safety question. The routine effluent releases were updated to include the Soddy-Daisy Falling Water Utility District and the new Tennessee River Miles. Since there are no regulatory requirements for population dosage. there is no unreviewed safety question.

SA-SE Number *WBPLEE-95-148-1*

Affected Documents	*Document Type*	*Safety Assessment Title*	*Implementation Date:*
DCN M-37645-A FSAR Change Pkg. 1416 1-PAT-8.3, R1 & 1-PAT-8.6,R1	FSAR	Turbine Trip Logic	3/2/96

Description of Change, Test, or Experiments

Revise the turbine trip logic so that the opening of a generator output breaker (e.g., PCB 5088 or 5064)) will cause a direct turbine trip.

Currently, Test Abstracts in FSAR Chapter 14 describe the Power ascension test program. These tests are based on the requirements of USNRC Reg Guide 1.68 Rev. 2. A strict interpretation of Reg Guide 1.68 paragraphs 5 1.1. and 5 n.n. requires Watts Bar to perform an additional trip from 100% power. Watts Bar's present design does not call for the automatic or direct trip of the turbine on loss of load feature and relies on the overspeed trip feature when the generator output breaker is manually opened. By installing the design modification that will directly trip the turbine upon opening the generator output breaker(s), then Reg Guide 1.68 paragraphs 5 1.1. and 5 n.n. can be combined, and only one 100% power trip test will be required.

DCN M-37645-A was issued to implement the changes described. Changes entail installation of new cable, timing relays and revision of the control circuit of the main turbine trip to add the extra feature that a manual trip of the generator output breaker will cause a direct turbine trip. Installation of an MG-6 relay and three Allen Bradley timing relays are made in relay board #9. FSAR Change Packages 1294 and 1416 are issued to change appropriate sections of Chapter 10 and 14, as are revision to Turbogenerator Control and Protection System (TGCPS) system description N3-47-4002.

Safety Evaluation Summary

The modification described in the DCN will then provide the same protection as defined for all postulated conditions that trip the turbogenerator, including a "manual" trip of the generator breakers (PCB 5088 or 5064). Therefore, the result of the change will be that if the generator output breakers are opened, an immediate turbine trip will result. If the breakers are opened below 50% power, no immediate reactor trip will occur, and the plant response will be the same as for a 50% load rejection. In this case the rapid closure of the turbine governor and intercept valves will cause the NSSS to behave essentially identically to the Turbine Trip without Reactor Trip analysis done as part of the P-9 setpoint implementation. Therefore, this change out will not cause a transient on the NSSS which is any more severe than those performed as part of the Turbine Trip without Reactor Trip below P-9 setpoint analysis done in 1982 for WBN.

Based on the detailed evaluation performed by Westinghouse relative to the NSSS and those provided by TVA relative to the actual hardware modifications, it is concluded that the design modification (DCN M-37645-A) to directly trip the turbine upon opening of the generator output breaker (PCB 5088 or PCB-5064) does not represent an unreviewed safety question as defined in 10CFR50.59, nor does it represent a change to the plant Technical Specifications.

SA-SE Number *WBPLEE-95-216-0*

Affected Documents	*Document Type*	*Safety Assessment Title*	*Implementation Date:*
TRM Change Pkg. 95-091 TRM Table 3.8.4-1 TRM Revision 1	Technical Requirements Manual and Bases	Correction of an Error in the TRM 3.8.4-1 Table	12/6/95

Description of Change, Test, or Experiments

This change may be made by performing a safety assessment only. There is no design change, FSAR is not impacted, and the change is to correct an error in the TRM table.

TS/TRM Change Package 95-091 deletes from TRM Table 3.8.4-1 circuit A45 on 125VDC Vital Battery Board I for control rod drive cooling unit 1C-A room diversion damper temperature control operator 1-TCO-30-90-A and circuit D7 on 125VDC Vital Battery Board II for SIS accumulator fill line isolation valve 1-FCV-63-24. This change is documentation only made in order to reconcile the supporting calculation WBNEEBMSTI080009 R8 and schematic drawings when the surveillance instruction was being developed to implement the TRM 3.8.4, it was discovered that a discrepancy existed between the TRM Table 3.8.4-1 and the design output documents. This change will make the calculation, schematic drawings, and TRM Table 3.8.4-1 agree.

Safety Evaluation Summary

The change is a nonsignificant change as it does not change the design, operational or performance requirements approved in the FSAR and Technical Specification and does not impact plant safety.

This change does not require new procedures to be issued or a revision to existing procedures. Surveillance Instructions 1-SI-99-300A and 1-SI-99-300B are being developed to monitor the equipment in TRM Table 3.8.4-1.

These components are not de-energized by safety injection signal and must be deleted from the TRM table to agree with the anlysis and schematic drawings. The actual loading changed, but remains below the protective device trip rating. Therefore, the TS/TRM change package does not decrease nuclear safety, and is not considered an unreviewed safety question.

SA-SE Number *WBPLEE-95-224-0*

Affected Documents	*Document Type*	*Safety Assessment Title*	*Implementation Date:*
DCN M-38269A FSAR Change Pkg. 1410	DCN	Customer Group Radio System	6/28/96

Description of Change, Test, or Experiments

DCN M38269 will remove the existing Customer Group (CG) Radio in the Control Building (CB) and will locate a new state of the art radio in the Turbine Building (TB). The existing antenna on the TB roof will also be removed and a new high gain antenna will be mounted in a more suitable location on the roof. The existing CG Radio is old and prone to numerous maintenance problems. Replacement parts are difficult to find because of the age of the radio and the fact that the vendor no longer manufactures radios. Reliability will be enhanced by replacing the old equipment with new equipment employing current technology. Similar equipment is also being installed at other TVA sites to improve the overall reliability of the valley-wide CG Radio System. This DCN will impact Radio System 253 and CB and TB Conduit and Cable Tray Systems 290 and 291. Also impacted is FSAR Section 9.5.2.4 which states that the CG Radio, the NSS Radio, and the Sheriff's Radio have "different locations so that it is unlikely that all would be out of service simultaneously." DCN M19796 has already relocated the NSS Radio to elevation 755' of the TB near the existing Sheriff's Radio. DCN M38269 will now relocate the CG Radio to the same area. The SAR statement is no longer true and must be deleted because the three radios will no longer be in separate locations. This SAR change is insignificant because the affected radios are not related to nuclear safety and there are no specific design, operational and/or performance requirements indicated. The SAR statement is simply stating that the radios are "unlikely" to fail simultaneously because they are in different locations. Further, there is no design requirement for these three radios to be in separate locations, and there is no benefit of redundancy to be gained from their separation because the Main Control Room has access only to the CG Radio.

Figure 9.5-19, which indicates the availability of communications systems for various postulated conditions, must be revised for editorial reasons and also for changes due to DCN M38269. The "Power System Operations VHF Radio" has for some time been known as the "Customer Group Radio" and this change will be made. This figure also shows the radio to be available for a fire in the control room. This should be changed to indicate "partial" survival of the system because the remote control unit in the control room obviously will not survive. Further, this figure indicates that the radio is not available for loss of all AC power for up to 3 hours. After DCN M38269 is implemented, the radio will be powered from a battery-backed Uninterruptable Power Supply (UPS) and will be available during this postulated condition. This change to the SAR figure denotes the enhanced availability of the CG Radio following a loss of power.

Safety Evaluation Summary

The design change is non-safety related equipment that will not impact the control, logic, or function of any safety related system, structure, or components. There are no accidents evaluated in the FSAR that affect this modification. The new radio will be operating at the same Radio Frequency (RF) and power range as the old radio and will not contribute any additional Electromagnetic Interference/Radio Frequency Interference (EMI/RFI) to the plant site. The relocation of the new antenna and it's increased directional characteristics will direct more of the RF energy toward the repeater on Roosevelt Mountain and less through the plant site. No additional RF energy will penetrate the concrete/steel construction of the plant buildings. Acceptance limits for the radio system are not discussed in the Tech Spec. Therefore the margin of safety can not be decreased. Therefore this modification will not increase the likelihood of interference with sensitive plant equipment and is not considered to be a unreviewed safety question.

SA-SE Number *WBPLEE-95-226-0*

Affected Documents	*Document Type*	*Safety Assessment Title*	*Implementation Date:*
DCN S-38582-A FSAR Change Pkg. 1404	FSAR	Equipment Hatch Doors	1/24/96

Description of Change, Test, or Experiments

Design Criteria WB-DC-40-60 and FSAR Section 3.8.1.1.1 provide requirements for equipment hatch doors installed in a sleeve in the Reactor Building. The steel sleeve forms an access through the Shield Building wall to the equipment hatch in the Containment Vessel. Included in the requirements are limit switches and visible and audible alarms in the main control rooms (MCR); however, it has been determined that no visible or audible alarms exist in the MCR. This DCN and FSAR Change Package removes the requirement for visible and audible alarms to be in the MCR.

Design Criteria WB-DC-40-59 provides requirements that door A216 leading into the Unit 1 Post Accident Sampling Facility (PASF) Room have an audible alarm within the MCR. DCN P-3000-B removed the audible portion of the MCR annunciation, however, did not address the design criteria. This DCN removes the requirement for the audible portion of the MCR annunciation for door A216. A visual alarm for this door exists in the main control room.

FSAR IMPACT DESCRIPTION

The FSAR was reviewed and Section 3.8.1.1.1 requires revision. This section identifies the requirement that there be a visible and an audible alarm in the main control room if the equipment hatch doors on elevation 757.0 are open. This hatch door will normally be opened only when the reactor is in shutdown, depressurized condition such that secondary containment is not required. It is not necessary to monitor the opening of the hatch due to the size and difficulty of opening this hatch. The FSAR does not discuss door A216 alarm features.

Safety Evaluation Summary

There are no unreviewed safety questions for this change. The secondary containment pressure boundary is maintained to support proper EGTS operation. Therefore, the Design Basis Accident safety analysis is not affected by this change.

SA-SE Number *WBPLEE-96-009-1*

Affected Documents	*Document Type*	*Safety Assessment Title*	*Implementation Date:*
DCN W-38644-B TS Change Pkg. 96-005 Bases Rev. 3 Closed 3/27/96	DCN	SIS Accum Tank Monitoring Channels	3/27/96

Description of Change, Test, or Experiments

This change makes the following changes to the SIS accumulator tank monitoring channels: high and low pressure alarm setpoints to be adjusted to actuated prior to reaching the Technical Specification limits, and level transmitters to be recalibrated to account for sense line reference leg high point elevation. The subject pressure loops are: 1-P-63-62, -62, -86, -88, -106, -108, -126, -128. The subject level loops are: 1-L-63-82, and -129. The level transmitter calibration change is required to correct reference leg high point m3easurement errors mad in the Demonstrated Accuracy Calculation.

The safety injection system (SIS) provides emergency core cooling and reactivity control for the following accidents: small and large break LOCAs, rod cluster control assembly (RCCA) ejection event, steam line break up to and including a MSLB, and steam generator tube rupture (SGTR) event. The SIS cold leg (CL) accumulators (four tanks per unit) provide a passive means to inject borated water into a depressurized reactor vessel through the cold leg injection lines. The borated water contents of the accumulator tanks are rapidly discharged by the nitrogen cover-gas pressure. The accumulator tank water inventory and nitrogen cover-gas pressure must be maintained within system limits to validate assumptions used in the accident analysis described in FSAR Chapter 15. This change ensures the tank level and pressure monitoring channels meet these specified system limits.

Each SIS accumulator tank uses two separate level channels and two separate pressure channels. These instrument channels are non-Class 1E. Also, each level channel provides a main control room indication (MCR) and actuates a high and low level annunciation in the MCR. Each pressure channel provides a MCR indication and actuates a high and low pressure annunciation in the MCR. This change may be performed in any mode of operation in accordance with operability requirements indicated in the technical specification.

Safety Evaluation Summary

This change is being made by performing a safety assessment only. This change does not affect any information presented in the FSAR or deviate from the description given in the FSAR. FSAR Section 6.3.5 discusses the SIS accumulator's safety function and design parameters. The SIS accum level and pressure indication and alarms are discussed. The alarm values are not specified in this section. FSAR Section 7.5 identifies the level and pressure channel indicator ranges. This change does not affect these indicator ranges. Therefore, this change does not affect the FSAR.This change does not involve any new procedures or revisions to existing procedures that affect system operational characteristics from that described in the FSAR or affect compliance with the Tech Spec.

The SIS accum tanks perform a safety function by injecting borated water into a depressurized reactor core to mitigate specific design basis accidents. The accum tank water inventory and nitrogen cover-gas pressure must remain within established safety limits in order to validate the accident analysis. This change does not affect any Tech Spec operability values and ensures the pressure and level monitoring loops provide adequate information to the unit operator. Therefore, this change does not decrease the margin of safety.

SA-SE Number *WBPLEE-96-013-0*

Affected Documents	*Document Type*	*Safety Assessment Title*	*Implementation Date:*
DCN M-38564-A	Fire Protection Program	Unnecessary Status Alarm Inputs to the Fire Detection System 13	4/15/96

Description of Change, Test, or Experiments

Change summary: DCN M-38564-A disconnects diesel generator building door position switch inputs from the fire detection system to disable the "door not closed" alarm. The alarm is no longer needed because these doors are normally closed and are equipped with automatic closers. A breaching permit with appropriate contingency procedures will be required for maintenance or operational activities requiring one of the doors to remain open for a significant period. The DCN disables the fire detection system alarm inputs for electric motor driven fire pump running and power available status. These alarms are duplicates of main control room annunciator alarms which are both audible and visual. The DCN also provides a different type input module for the diesel fire pump status inputs to ensure that the pump running status will not mask other supervisory alarms.

SAR impact: There is no direct impact to the FSAR, but the Fire Protection Report must be revised to delete 'door position' switches from the list of fire detection system inputs. The description of the MCR alarms for the Electric and Diesel Fire Pumps in the Fire Protection Report remains valid.

Safety Evaluation Summary

DCN M-38564-A does not affect any FSAR accident analysis or equipment. The fire detection system door position status inputs are not required because the normally closed doors have automatic closing mechanisms and breaching permits are required if doors are to be kept open. The fire pump status alarms are not needed because they are duplicates of main control room annunciator alarms which are both audible and visual. The changes do not affect the fire detection or suppression functions; they only affect status monitoring. The Fire Detection System is not required for mitigation of any FSAR Chapter 15 Design Basis Accidents. Therefore, based on evaluation of effects, it is concluded that the proposed change is acceptable from both nuclear safety and fire protection perspectives, and no unreviewed safety question exists.

SA-SE Number *WBPLEE-96-018-0*

Affected Documents	*Document Type*	*Safety Assessment Title*	*Implementation Date:*
DCN S-38679-A	Fire Protection Program	Manual Actions Required for Fire Safe Shutdown (FSSD) Following a Fire	3/11/96

Description of Change, Test, or Experiments

Change summary: DCN S-38679-A changes the method of causing the reactor building lower containment cooling temperature control valves and inlet dampers to fail open as required for safe shutdown following an Appendix R fire. The previous method was to remove control air by isolating and depressurizing the reactor building non-essential control air header. The revised method is to remove power from the associated solenoids and modulating controllers by opening 120VAC breakers and removing 125VDC fuses. This DCN is the design output basis for a corresponding change to AOI 30.2, "Fire Safe Shutdown".

SAR impact: There is no direct impact to the FSAR, but the Fire Protection Report must be revised to change the description of the manual actions credited to ensure the containment cooling TCV's and TCO's are open within 2 hours when containment cooling is required following a fire.

Safety Evaluation Summary

DCN M-38679-A does not modify plant equipment or normal operating requirements or methods. The proposed change is procedural and only affects post fire safe shutdown abnormal operating procedures. The required safe shutdown function, ensure containment cooling by failing open temperature control valves and inlet dampers, is not changed. Loss of control air or electrical power results in the same valve/damper failure position. Power removal can be affected in the same or less time than control air removal. Access/egress/emergency lighting to the areas required for the revised manual action has been assured because other manual actions already require access/egress to these areas. Therefore, based on evaluation of effects, it is concluded that the proposed change is acceptable from both nuclear safety and fire protection perspectives, and no unreviewed safety question exists.

SA-SE Number *WBPLEE-96-022-0*

Affected Documents	*Document Type*	*Safety Assessment Title*	*Implementation Date:*
DCN M-38633-A FSAR Change Pkg. 1420	FSAR	EHC Low Pressure Turbine Trip and EHC Lo Tank Level Trip	5/18/96

Description of Change, Test, or Experiments

The subject DCN eliminates the EHC Low Pressure Turbine Trip and the EHC Lo Tank Level Turbine Trip in order to prevent spurious trips from these sources.

The EHC low pressure turbine trip signal is currently an output of 1-PS-47-020. This pressure switch also has an output to the annunciator system (panel 1 M-4) and the plant computer. The trip function will be deleted. Annunciation for this signal is currently window 73A on XA-55-4C, "EHC Fluid Pressure Low", which is in the "Turbine Trip First Out" lamp box. Since the current configuration is associated with the turbine trip panel, a deletion of the trip signal should be accompanied by a deletion of the annunciator signal to the panel. Therefore, the annunciator function of the switch will be relocated to XA-55-2A (panel 1 M-2) alongside other turbine control annunciator windows. The alarm for the pressure switch is being relocated to the same window. Reflash capability is being added and the printer legend is being changed so the origin of the alarm can be determined. The only change to the computer is a nomenclature change reflecting the change from the trip function.

The EH Tank Fluid Low Level turbine trip signal currently an output of 1-LS-47-007A/B, also has an output to the first out annunciator system (Panel 1 M-4), the plant computer, as well as control functions. Annunciation for this signal is currently at window 72A on Lamp Box XA-55-4C, "EHC Tank Level Lo". Since this alarm is on the turbine trip panel, the annunciator function of the switch will also be relocated (Panel 1 M-2) XA-55-2A, window 24B and will be a second input to the existing window (as decided by human factors engineering and the plant) as with the low pressure switch, the only change for the output to the computer is in the switch nomenclature. As part of the turbine trip deletion, the EHC pump trip for low EHC fluid level is also being deleted.

Safety Evaluation Summary

This modification is limited to the turbine trip and EH pump trip functions and provides for a nomenclature change to the computer termination and I/O list. Review of the detailed changes leads to the SE conclusions that this modification is safe and does not constitute an unreviewed safety question. Physical wire changes related to 1-LS-47-7A/B, which currently actuates Relay 86/LFT, will be performed in JB 673. Relay 86/LFT, a mechanically held, latching relay with manual reset, presently provides contacts for turbine and EH pump trip, EH pump enable, and associated annunciation, and plant computer input functions. Relay 86/LFT provides enable and trip signals for both EH pumps as well as turbine trip signals in the event that the EH fluid level drops below the depleted level setpoint. This modification electrically disconnects wiring to those relay contacts associated with EH pump and turbine trip. The contacts providing permissive signals to the EH pumps and computer input signals will be unaffected.

Considering the operational actions relative to the alarm aspects of the change, it must be understood that the setpoints of the two pressure switches being annunciated are different. The switch for the existing alarm at window 24B has a setpoint of 1600 psig. The alarm setpoint for the switch that previously tripped the turbine is 1350 psig. If an alarm is received at this window the operator by procedure will check the printer to determine which switch initiated the alarm. If the alarm is the 1600 psig setpoint, the operator will initiate the actions described in item 20 of the SA Checklist, and as captured in the system description, i.e., verify proper EH Pump operation, determine it there is any leakage in the EH fluid system. If, subsequently, the 1350 psig alarm comes in, there is sufficient cause to suspect that this is not a spurious alarm and the operator will continue further with the actions to start to remove unit load. If the Main Throttle/Intercepts or Reheat Stop Valves start to close the operator will take the final action dictated, i.e., trip the turbine. In this sequence of events the only change to the system operational design is that the unit is tripped manually instead of automatically. The impetus for the turbine trip is still the same switch setpoint, i.e., 1350 psig. Should the operator identity that only the 1350 psig alarm is received without ever receiving the 1600 psig alarm, this alarm condition without the symptoms of Main Throttle and Reheat Stop Valves starting to close could be potentially construed as a spurious, with actions to investigate for proper EH pump operation, system leaks, and the initiating instrumentation itself. On the other hand, observation of valve closure regardless of indication of the 1600 psig signal first would be sufficient cause for operator action to trip the turbine.

The potential was investigated for EH fluid pressure decreases (small system leak as compared to a sudden depressurization) where throttle/intercept/reheat control/stop valve asymmetrical closure with "Bottling" of steam in any one of the turbines could cause undue stresses to the turbine shaft/rotor/blades. In view of the Human factors approach without the automatic EHC low pressure trip, potential for decrease to the original 1350 psig trip setpoint, and the operator actions outlined preceding this point (i.e., subsequent to 1600 psig alarm) and as a result of this second alarm, the pressure decrease should result in closing the throttle and intercepts sooner than the reheats due to the relative steam pressure/valve spring force considerations. This should then lead the operator to trip the turbine before reaching the condition of concern. In addition, a review was requested of Westinghouse (site/Orlando) for experience to address TVA concerns that throttle/intercept/reheat control/stop asymmetrical action could occur some manner due to a slow pressure decrease at some point in the EH fluid tubing to the HP/LP turbine control valves. The review indicated no information was found relative to this concern. Westinghouse advised that generally, a fast pressure decrease leading to dumping of all the valves (not unlike a normal turbine trip at 1350 psig) would not be unlike the operator taking manual action at this same point by procedure. Further, considering closure of one of the pair of the respective turbine control and/or stop valves relative to one of another pair, due a postulated EH system leak closest in proximity to the faster closing valves, has also not been found to occur in experience, however for this condition to cause undue stresses on the turbine equipment is not expected.

Review of detail and operational changes as well as EHC asymmetrical valving considerations leads to the SE conclusions that this modification is safe and does not constitute an unreviewed safety question. In addition, this modification will be made with the turbine out of service; therefore, from this aspect as well there will be no adverse affects which may cause turbine/reactor trips and present a challenge to plant safety systems.

SA-SE Number *WBPLEE-96-023-0*

Affected Documents	*Document Type*	*Safety Assessment Title*	*Implementation Date:*
DCN M-38674-A FSAR Change Pkg. 1413	FSAR	Nuisance alarms in the Main Control Room	3/7/97

Description of Change, Test, or Experiments

This change, DCN M-38674-A, removes plant door position indication for status panel, 1-XA-55-23A, located in the main control room (MCR) and places the electrical interlock feature under Unit 1 configuration control for door pairs A105/A130, A132/A133, and A214/A192. These door pairs are associated with Unit 2 structural features, however, they are part of the Unit 1 Auxiliary Building Secondary Containment Enclosure (ABSCE). Also, the horn/chime is reconnected to provide an audible alarm associated with the remaining Cntmt Air Lock Doors and the Cntmt Barrier Divider Hatch position indication.

The door position indication panel monitors doors that allow entrance to various equipment rooms and to general plant areas. This door position information is not needed for unit operation personnel for safe operation of the plant. Doors that are used for normal plant egress/ingress and that provide pressure boundary integrity for the Secondary Containment Enclosure (Auxiliary and Shield Building Environment) are maintained closed by use of electrical interlocks for the air lock pair. The remaining pressure boundary doors do not use electrical interlocks because of infrequent egress/ingress through them during an Auxiliary Building Isolation event. All Auxiliary Building Secondary Containment Environment (ABSCE) boundary doors are required to be identified by the use of signs located near the door. Also, personnel radiation exposure control is administered by RADCON as part of the Radiation Work Permit (RWP).

FSAR IMPACT DESCRIPTION

FSAR Section 6.2.3.2.1 discusses the Auxiliary Building Secondary Containment Enclosure and Section 6.2.3.3.1 discusses the Shield Building Secondary Containment Enclosure. Statements are made in both sections associated with local and Main control room alarms to detect if both air lock doors are ever opened simultaneously. These alarms are not required in accordance with design basis documents. This FSAR Change Submittal reflects this design feature.

Safety Evaluation Summary

This change, DCN M-38674-A, does not affect any FSAR evaluations (accident analysis or equipment malfunction failures) previously performed. No new accidents or equipment malfunction failures are created. Technical Specification is not affected. This change is in compliance with safety classification requirements specified in design basis documents. Therefore, on the basis of the evaluation of effects, it is concluded that the proposed change is acceptable from a nuclear safety standpoint and no unreviewed safety question exists.

SA-SE Number *WBPLEE-96-026-1*

Affected Documents	*Document Type*	*Safety Assessment Title*	*Implementation Date:*
DCN S-38673-B FSAR Change Pkg. 1417 S1	FSAR	Turbogenerator Control and Protection System	4/16/96

Description of Change, Test, or Experiments

The high turbogenerator vibration turbine trip signal is currently an output of Vibration No. 1 (XX-47-121) and/or Vibration No. 2 (XX-47-131), which are for the turbine and generator bearings and exciter bearings. Currently, with handswitch 1-HS-47-120 in the normal position, turbine and generator vibration of 14 mils and Exciter vibration of 14 mils can trip the turbine automatically, i.e., subsequent to MCR alarm at 7 mils warning the operator that this equipment is beyond the normal 4 mils specification. Placing hand switch 1-47-120 in the "cutout" position will disable the automatic trip function and first out annunciator indication and alarm window 73B (Panel 1 M-4) which would normally be actuated at 14 mils. However, the alarm function alarm at 7 mils (window 23A on panel 1 M-2) remains unchanged . Analog inputs that represent turbine-generator bearing vibration will still be input to the P-2500 computer and data linked into ERFDS. The P-2500 performs two functions related to each input. An alarm is generated by the P-2500 at 7 mils and the graphical user display changes to magenta above 7 mils. Although having the 1-HS-47-120 in the "cutout" position prevents an annunciator alarm (as well as the automatic trip) at 14 mils, the P-2500 analog inputs are scaled beyond 14 mils and will allow trending.

Implementation of the change will require a revision to TVA WBN FSAR sections 10.2.2.,10.2.4 and 3.5.1.3.1 to delete reference to this trip function. In addition, comparable sections of the Turbo generator Control and Protection System Description N3-47-4002 will be made, to clarify that the automatic trip and alarm (14 mils) has been disabled, and that the plant operators will be responsible for monitoring the turbine and generator, and exciter vibration beyond the limits of the MCR 7 mil alarm i.e., trip is manual at the operators discretion.

Safety Evaluation Summary

Review of the detail changes leads to the SE conclusions that this modification is safe and does not constitute an unreviewed safety question. The vibration trip set at 14 mils was intended to minimize damage that could occur to the turbine unit should vibration increase above this point. This trip is not needed to shutdown the reactor or protect the health and safety of the public for radiological accidents. The alarm at 7 mils which still exists, is to warn the operator that vibration has increased (and could continue to increase to the automatic trip setpoint if not disabled as intended) beyond the normal specified 4 mils. If vibration increases occur in a slow progression or a step change occurred sufficient to create an alarm at 7 mils it would then be cause for close monitoring of the signal if the automatic trip is disabled. If vibration level continued to increase toward 14 mils, then this would allow for a possible planned corrective action or administrative decision to remove the unit from service versus an automatic unit trip. In considering disabling the automatic trip function already designed into the plant, a review of the industry in 1995 indicated that only 2 of 13 plants with Westinghouse turbine generators still had the 14 mil vibration trip in service. The other 11 plants had changed the trip to a second level alarm allowing administrative controls on the trip. In 1996 only one plant of the 13 originally surveyed nuclear plants are operating with the 14 mil vibration trip in service. In addition the TVA Westinghouse fossil units do-not use the vibration trip feature but instead use it as a second level alarm. At WBN the high vibration trip is a one for one signal which is generated from 11 different bearings, whereby any of these signals can trip the unit.

SA-SE Number *WBPLEE-96-029-0*

Affected Documents	*Document Type*	*Safety Assessment Title*	*Implementation Date:*
DCN S-38722-A	Fire Protection Program	Auxiliary Control Air System Analysis	3/12/97

Description of Change, Test, or Experiments

Change summary: DCN S-38722-A changes the method of operating the Motor-driven Auxiliary Feedwater Level Control Valves (LCVs)/Pressure Control Valves (PCVs) and Steam Generator PORVs for safe shutdown following an Appendix R fire in a number of fire zones. The previous method was to locally operate these valves for fires in "all" plant locations. The revised method is to perform local operation only for those plant locations where remote operation from the main control room (Auxiliary Control Station for MCR fires) is not available.

Additional changes due to this DCN include a revision to the Appendix R compliance strategy in Fire Zone 772-A12 to use Steam Generators 3 and 4 for cooldown in lieu of Steam Generators 1 and 2. Also, manual actions (611 and 613) to address the potential for a spurious SI actuation signal were deleted in Fire Zones 729-A12, -A14, -A16 and 763.5-A1 since the cables associated with the instrumentation that could cause such a signal were determined not to be exposed to the effects of a fire in these zones (i.e., conduits are embedded).

Additionally, 737-A5 was divided into three new subzones (737-A5S, N. M) and re-analyzed. This was necessary to address separation between redundant components/cables that are located in 737-A5. The compliance strategy now credits SG 3 in addition to SGs 1 and 4 to achieve cooldown for fires in this room.

This DCN is the design output basis for a corresponding change to A01 30.2, "Fire Safe Shutdown".

SAR impact: There is no direct impact to the FSAR, but the Fire Protection Report (FPR) must be revised to change the requirement for the performance of the manual actions when remote operation from the Main Control Room is not available following a fire. The FPR must also be revised to add/delete other zone-specific actions based on Appendix R re-analysis.

Safety Evaluation Summary

DCN S-38722-A does not modify plant equipment or normal operating requirements or methods. The proposed change is procedural and only affects post fire safe shutdown abnormal operating procedures. The required safe shutdown function, to ensure cooldown capability, is not changed. Therefore, based on evaluation of effects, it is concluded that the proposed change is acceptable from both nuclear safety and fire protection perspectives, and no unreviewed safety question exists.

SA-SE Number *WBPLEE-96-031-0*

Affected Documents	*Document Type*	*Safety Assessment Title*	*Implementation Date:*
FSAR Change Pkg. 1418 WB-DC-30-8	FSAR	Safety Injection System Cold Leg Accumulator Tank Level Measurement System	4/3/96

Description of Change, Test, or Experiments

This change, FSAR Change Package No. 1418, updates FSAR Section 6.3.5.4 and Table 7.5-2 (Sheet 5 of 18) to describe the Safety Injection System (SIS) Cold Leg Accumulator (CLA) Tank level measurement system. Specifically, Section 6.3.5.-4 states that two level channels are provided for each tank. The installed configuration uses one thermal dispersion type monitor (supplied by FCI) and one differential pressure transmitter (supplied by Rosemount). This change deletes the term "identical". Also, Table 7.5-2 (Sheet 5 of 18) identifies the CLA indication range to be 7632 to 8264 gallons. The design range of the CLA Tank level channels is 7450 to 8080 gallons. This change revises the FSAR to reflect the design range. Also, this change revises Design Criteria, WB-DC-30-7, "Post Accident Monitoring Instrumentation" to reflect the proper range.

Safety Evaluation Summary

This change does not affect any FSAR evaluations (accident analysis or equipment malfunction failures) previously performed. No new accidents or equipment malfunction failure are created. Technical Specification is not affected. This change is in compliance with safety classification requirements as specified in design basis documents. The Safety Evaluation Report (SER) and Supplemental Safety Evaluation Reports (SSERs) Nos. 1 through 20 have been reviewed and no impacts were found. Therefore, on the basis of the evaluation of effects, it is concluded that the proposed change is acceptable from a nuclear safety standpoint and no unreviewed safety question exist.

SA-SE Number *WBPLEE-96-032-0*

Affected Documents	*Document Type*	*Safety Assessment Title*	*Implementation Date:*
DCN M-38637-A	DCN	Turbine Trip and Alarm Logic Modified	4/16/96

Description of Change, Test, or Experiments

The proposed plant modification in the subject DCN M-38637-A revises the turbine trip and alarm logic initiated by the stator cooling water system process variables. The current logic initiates a turbine trip when a single differential pressure (dp) switch (one/one logic) senses low dp water pressure across the generator stator coil or high temperature at the stator coil cooling water outlet. This modification adds two differential pressure switches (the third dp switch 1-PDS-35-120B already exists), and three new temperature switches and wells to sense the process variables to provide a two/three logic.

The subject DCN also makes corrections to the Turbine Generator Control and Protection System description to capture the proposed plant modification, i.e., the change from the current logic which initiates a turbine trip when a single differential pressure switch or temperature switch (one/one logic) senses low differential water pressure across the generator stator cooling water coils or high temperature at the stator cooling water coil outlet, to a two/three logic. The setpoints currently associated with stator cooling coil dp switch 1-PDS-35-120B and stator cooling water discharge temperature switch 1-TS-35-104, will remain unchanged, i.e., upon detecting a differential pressure of 17 psi decreasing for two/three dp switches or 194 °F increasing for two/three temperature switches a turbine trip is initiated if the trip signal is sustained for 45 seconds and if the plant power level is at 15% or above. Also, appropriate corrections are made to related Vendor Technical Manuals. This modification and post modification test of each two/three logic gate for the differential pressure and temperature switches to perform as designed and pressure test for piping (new thermowells), will be made with the turbine out of service.

Safety Evaluation Summary

FSAR section 10.2 discusses the turbine generator. However, the turbine trip instrumentation involved in the proposed plant modification is not specifically addressed. This change does not add or delete any existing turbine trip feature, but only enhances the reliability of the existing feature thereby reducing the potential for spurious turbine trips and any subsequent potential reactor trips greater than 50% load reduction above P-9). This change does affect FSAR Figures 10.2-1 and 10.2-3 by the addition of the temperature and differential pressure switches required to implement the two/three logic configuration.

After a detailed review, it was determined that this proposed activity will in effect decrease the probability of a Condition II turbine trip event occurring (see FSAR section 15.2.7), i.e., less spurious turbine and subsequent potential reactor trips. In addition, if less spurious turbine trips occur from this source, then effectively there should be a decrease in the probability of a malfunction of secondary as well as primary equipment. The total transients for turbine trips throughout the plant life considered in the initial design of the turbogenerator itself, and more specifically as would affect the primary plant due to spurious trips of the turbine above 50% power do not specifically define equipment usage resulting from these specific trips, but does provide an accumulative for all trips over the expected life of the plant as delineated in the FSAR and Turbogenerator Control and Protection System description. Therefore, as there should be less spurious turbine trips from these initiators, there should be no change in the consequences (increase of an accident or malfunction of equipment to the secondary as well as the primary plant) than has already been set forth for a turbine trip from any initiator as defined in the FSAR (e.g., FSAR section 15.2.7). Also, since the setpoints for the two/three logic are not changed from that used for the one/one logic, and the results of the logic initiation of turbine trip has not changed (e.g., auto stop oil pressure logic), there is effectively a decrease in the probability of an accident or malfunction of equipment of a different type from occurring. Lastly, the activity does not reduce the margin of safety as defined in the Technical Specification or respective bases. Effectively, a reduction in potential spurious trips, will increase the margin of safety. Further, the integrity of the core is still ultimately maintained by operation of the Reactor Protection system, and the DNBR will still be maintained such that it will still not go below the design basis limit. The change does not impact the operation of the RPS system, and there will be no increased probability of cladding damage and increase in release of fission products to the RCS as a result of the intended modification.

SA-SE Number *WBPLEE-96-037-1*

Affected Documents	*Document Type*	*Safety Assessment Title*	*Implementation Date:*
DCN M-38766-A FSAR Change Pkg. 1424	FSAR	Setpoing for Annulus Pressure Controllers	4/18/96

Description of Change, Test, or Experiments

Train A and B annulus pressure controller PDIC-65-80 and PDIC-65-82 controls the position of EGTS Train A and B modulating dampers PCO-65-80, -88 and PCO-65-82, -89, respectively, in response to changing vacuum in the annulus. Currently, the set point of the controllers is -1.048 inches of water differential. During system start up subsequent to a LOCA, the time response of the control loops allow the annulus differential pressure to increase to -0.642 inches of water. Train A and train B bistables are set to close the isolation valves in the discharge lines for the running train of EGTS (i.e., the train in "A-auto" status), and to open the isolation valves in the discharge lines for the train of EGTS in standby when the annulus differential pressure increases to -0.812 inches of water; i.e., the bistable logic perceives a failure has occurred in the operating train. With assumption of a single failure in the standby train of EGTS, no EGTS flow path will be available. As required by the EGTS system description, at least one train of EGTS must be available to insure off site and main control room radiation dose limits remain within allowable limits subsequent to a design basis accident. DCN M-38766-A changes the setpoints of the controllers from -1.048 inches of water to -1.45 inches of water. This setpoint change will result in a post LOCA peak annulus differential pressure of -1.064 inches of water, thereby preventing switch over of the EGTS trains.

Revision 1 is issued to correct minor typographical errors. This revision does not change technical content or conclusions.

Safety Evaluation Summary

An USQ does not exist for the DCN M-38766-A to change the controller set point from -1.048 inches of water to -1.45 inches of water because the probability or consequences of an accident or equipment malfunction is not increased by the change. In addition, the change does not create the possibility of an accident or malfunction of a different type than any previously evaluated in the SAR and the margin of safety defined in the Tech Specs is not reduced since the maximum annulus differential pressure subsequent to a LOCA remains more negative than -0.61 inches of water as required by the Tech. Specs

SA-SE Number *WBPLEE-96-042-0*

Affected Documents	*Document Type*	*Safety Assessment Title*	*Implementation Date:*
DCN M-38800-A FSAR Change Pkg. 1425	DCN	Control Logic for HDT Discharge Control Valve	4/20/96

Description of Change, Test, or Experiments

This change, DCN M-38800-A, revises the No. 3 Heater Drain Tank (MDT) Pump runout protection logic associated with discharge control valve, 1-LCV-6-106A. This change provides the following logic to position the discharge control valve to a preset position (~30% open) upon meeting the following conditions:

· Any No. 3 HDT Pump not running
-AND-
· Unit turbine load > 85% (1-PS-47-13E)

A time delay relay is used to ensure the condition exists for a preset time period (~10 sec) in order to eliminate spurious control signal actuation due to potential switch contact chatter. The existing control valve logic uses differential pressure (dp) switches to indicate excessive pump flow. This excessive flow condition was caused by a postulated trip of a No. 3 heater drain pump with the unit turbine load above 85%. This change simplifies the discharge control valve logic associated with the No. 3 heater drain pump runout protection.

FSAR Impact
FSAR Chapter 10, Section 10.4.10.3 will be updated to more accurately reflect this design change as follows:
"During unit operation above 85% turbine load, a trip of a No. 3 heater drain pump will cause excessive flow in the two remaining operable pumps. This excessive flow condition may cause the operable pumps to experience inadequate NPSH. Upon detection of this condition, the No. 3 heater drain tank discharge level control valve is automatically positioned to a preset value to limit discharge flow and provide for pump protection."

Safety Evaluation Report (SER) June 1982 (including supplements 1 through 20) are not impacted because a review indicates the Heater Drains and Vents System is not addressed.

Safety Evaluation Summary

This change, DCN M-38800-A, does not affect any SAR evaluations (accident analysis or equipment malfunction failures) previously performed. No new accidents or equipment malfunction failures are created. Technical Specification is not affected. This change is in compliance with safety requirements as specified in design basis documents. Therefore, on the basis of the evaluation of effects, it is concluded that the proposed change is acceptable from a nuclear safety standpoint and no unreviewed safety question exist.

SA-SE Number *WBPLEE-96-056-3*

Affected Documents	*Document Type*	*Safety Assessment Title*	*Implementation Date:*
DCN W-38729-A DCN F-39002-A (Closed 9/17/96) FSAR Change Pkg. 1431	DCN	Radiation Monitor Malfunction Alarms with Ratemeters	12/2/96

Description of Change, Test, or Experiments

DCN W-38729-A modifies the configuration of the malfunction alarms for radiation monitors with ratemeters on main control room panel 1-M-30 such that the alarms will reflash on a single window on panel 0-M-12. Previously, the malfunction alarms were tied to a single window on panel 1-M-30 which did not have reflash capability so that it was not possible to recognize the initiation of subsequent alarms based on the annunciation window. The locations of the malfunction alarms are described in FSAR Table 11.4-3.

In order to provide reflash capability, the malfunction alarms from trained ratemeters must be tied to dedicated isolation relays separating the class 1 E radiation monitoring equipment from the non-qualified main control board annunciator. Previously, all alarms of one train were tied together to energize the coil of one separation relay per train.

The worst case result of the addition of the relays is a slight increase in probability of relay fault resulting in the trip of a breaker in the 120VAC vital board, subsequently resulting in the loss of one train of radiation monitoring equipment located on 0-M-12 and 1-M-30. A trip of the breaker could also occur due to inadvertent shorting while implementing the modification. The affected equipment includes Tech Spec radiation monitor loops 0-RE-90-102 and 125, and 1-RE-90-106, 271, and 273, for Train A; or 0-RE-90-103, and 126, and 1-RE-90-112, 272, and 274 for Train B. The affected equipment also includes ODCM monitor loops 0-RE-90-133, and 140 for Train A; and 0-RE-90-134 and 141 for Train B. These radiation monitors perform the following functions: isolation of the auxiliary building ventilation system following a fuel handling accident, isolation of the control room following a steam generator tube rupture or a fuel handling accident, primary indication to the control room operator of potential LOCA, and RCS leakage detection, and detection of a potential radioactive leakage path to the environment.

Safety Evaluation Summary

The addition of the dedicated relays does not result in different failure modes than existed with the single interposing relay and thus does not introduce a new equipment malfunction. This change does not interface with other plant equipment such that there is any potential for a different accident.

The consequences of an accident are not increased because the potential for losing two trains of radiation monitoring equipment simultaneously is not created and thus is not credible. The indication functions will be lost for a single train in the event of the breaker trip. The ESFAS functions are fail safe and will actuate on loss of power.

The potential for tripping the breaker and affecting a broader scope of radiation monitoring equipment than that involved in the change is minimized during implementation of the change by disconnecting the power supply jumper to the equipment being modified Additionally, DCN 38729-A is staged such that it is possible to allow work in any mode. This staged approach allows the modification to be implemented within the LCO boundaries of the Technical Specification. The DCN also allows implementation in Mode 5 or 6 in its entirety (not staged) with fewer LCO restrictions required.

The addition of reflash capability for malfunction alarms for radiation monitoring equipment represented on panel 1-M-30 increases the operator's ability to locate an inoperable loop, of which 1-RE-90-271 through 274 are of primary interest due to their R.G. 1.97 Category 1 Type A classification. This change thus improves the ability to be prepared to limit the consequences of a LOCA.

SA-SE Number *WBPLEE-96-082-1*

Affected Documents	*Document Type*	*Safety Assessment Title*	*Implementation Date:*
DCN M-38746-A FP06	Fire Protection Program	Thermal Fire Detectors Installed in North and South Main Steam Valve Rooms	6/4/97

Description of Change, Test, or Experiments

This change, DCN M-38746-A is composed of two independent parts. The first part is to abandon in place thermal fire detectors installed in the North and South Main Steam Valve Rooms (MSVR) (729-A1 and 729-A2). This is accomplished by disconnecting cables FE7337, FE7341, FE7342, and FE7346 in panel 0-L-624. These cable feed loops containing detectors TS-13-309A, B. D, E, F. G. H. J. and K (zone 332) in the North Main Steam Valve Room and detectors TS-13-310A, B. D, E, F. G. H. J. K and L (zone 333) in the South Main Steam Valve Room. Disconnected cables are to be sealed and secured for protection. Existing cables completing the loops between detectors will remain connected in place.

These detectors are spuriously actuating and staying in alarm due to the high temperature and humidity conditions in these rooms during plant operation. These detectors are also located on or near the ceiling in an area that is congested and relatively inaccessible making maintenance and testing difficult.

In an effort to determine the reason the detectors were installed initially, the system description for the fire detection system N3-13-4002 R0 was reviewed and compared to the current N3-13-4002 R2. Section 2.0 Design Criteria for both revisions indicate that "the fire detection system shall be installed in all areas of the plant that contain or present an exposure fire hazard to safe shutdown or safety-related systems or components and where property loss is unacceptable". It is conjectured that a conservative approach was taken by the original designer and installed the detectors in the main steam valve vaults because safe shutdown equipment is located within the area. This change, DCN W-38746-A, meets the above design criteria in that there is an "insignificant" exposure to fire hazards as explained in the first bullet of the preceding paragraph. There is no NFPA or NRC requirement to have detectors in this type room and removal does not violate codes nor design requirements.

The second part of this DCN is to disconnect relays PS-305 and PS-500 in relay panel 1-R-79 to prevent automatic start of the electric fire pumps due to alarm inputs from detectors in the Engineering and Quality Building (EQB) and the Modifications Building (MDB). These relays will remain in place. The change is required to prevent undesired starts of the electric motor driven pumps due to unwanted alarms in these two non-seismic, non-quality related buildings. These starts cause unnecessary stress and degradation to the motors.

Safety Evaluation Summary

Elimination of the detectors is justified because of the following:

o The combustible loading within the MSVRs is classified "Insignificant" in calculation EPM-DOM-012990 R5, Combustible Loading Data (CDL) Summary, pages 203 and 204, and in Fire Protection Report, Part VI, Sections 3.16 and 3.17. Combustibles in these rooms consists of lube oil in the valves and plastic associated with electrical panels and boxes and lights.

o The walls of the MSVRs are reinforced concrete.

o No cable trays are located in the MSVRs.

o These rooms are not protected by the fire suppression system, thus the detectors are used for alarm only.

o Fire protection systems and features at WBN are not assumed to be operable to mitigate the consequences of a Design Basis Accident or plant transient (Fire Protection Report, Part II, Section 14.0).

Disconnecting the relays is justifiable because the suppression system piping is pressurized by the Raw Service Water (RSW) system and will supply water to the sprinkler heads if needed. Low header pressure will cause an automatic isolation from the RSW system and automatically starts the diesel fire pumps to maintain pressure in the piping. Also, upon alarm by detectors in these two buildings and verification of an actual fire, control room operators are procedurally required to start the electric fire pumps.

These changes do not affect any FSAR evaluations (accident analysis or equipment malfunction failures) previously performed. No new accidents or equipment malfunction failures are created. The Technical Specifications are not affected. This change is in compliance with safety classification requirements as specified in design basis documents. Therefore, on the basis of the evaluation of effects, it can be concluded that the proposed changes are acceptable from a nuclear safety standpoint and no unreviewed safety question exists.

SA-SE Number *WBPLEE-96-084-0*

Affected Documents	*Document Type*	*Safety Assessment Title*	*Implementation Date:*
DCN S-38919-A AOI-30.2	Fire Protection Program	Manual Operator Action Requirement Changes	3/20/97

Description of Change, Test, or Experiments

Change summary: DCN S-38919-A changes fire safe shutdown manual operator action requirements to correct an error discovered during re-analysis using the WBN specific version of INDMS - SSAS (Integrated Nuclear Data Management System - Safe Shutdown Analysis System) software. Previous analysis directed closure of 1-FCV-74-21-B to block potential RWST (Refueling Water Storage Tank) drainage to the reactor building sump for a fire in room 772-A2 (east). Re-analysis shows that fire damage to cables in 772-A2 (east) could prevent main control room operation of 1-FCV-74-21-B. Both analyses indicated that 1-FCV-63-1-A would be free of fire damage and operable from the main control room, but the manual action to close 1-FCV-63-1-A was not specifically identified in the Manual Action Calculation.

Several minor changes to manual action requirements are also included in DCN S-38919. These minor changes are considered to be simplifications, enhancements, added conservatism, completeness, or consistency changes because the existing analysis and operating procedure were fully adequate. These changes are not safety requirements but are included to reduce or simplify manual actions, clarify existing actions, or to provide operational flexibility.

A listing of available diagnostic instruments associated with the credited safe shutdown path is provided in DCN S-38919. Previously a listing of all available diagnostic instruments associated with FSSD systems and components was provided. The change to provide only the success path instruments is intended to simplify procedures by not listing instruments which really don't provide usable information.

SAR impact: There is no direct impact to the FSAR, but the Fire Protection Report (FPR) must be revised to change the reflect the current manual operation requirements.

Safety Evaluation Summary

DCN S-3891 9-A does not modify plant equipment or normal operating requirements or methods. The proposed change is procedural and only affects post fire safe shutdown abnormal operating procedures. Other than correction of one erroneous manual action, the required safe shutdown functions are not changed. The plant's capability to accomplish the necessary fire safe shutdown functions is enhanced by the proposed manual operator action and available instrumentation list changes. Therefore, based on evaluation of effects, it is concluded that the proposed change is acceptable from both nuclear safety and fire protection perspectives, and no unreviewed safety question exists.

SA-SE Number *WBPLEE-96-093-0*

Affected Documents	*Document Type*	*Safety Assessment Title*	*Implementation Date:*
DCN W-38897-A FSAR Change Pkg. 1440	DCN	Main Control Room Annunciators "Darkboard" Issues	8/26/96

Description of Change, Test, or Experiments

The Control Room Design Review (CRDR) project identified various annunciators which violated the goal of having all main control room (MCR) annunciators meet the "darkboard" criteria during 100% power operations. All "darkboard" issues were not identified during the CRDR project, therefore this DCN addresses an additional issue. This change, DCN W-38897-A, revises the setpoint for the Cask Decontamination Collector Tank (CDCT) high level alarm from approximately 74% tank level to approximately 92% to allow the maximum amount of water to be released to the river during a permitted discharge. This alarm originates at panel 0-L-725 and cascades to annunciator window 16 on PNL 0-L-2D and window 161-F on PNL 1-M-15 in the MCR.

This DCN also addresses a second reason why window 161-F does not meet the "darkboard" criteria. Windows 13 on 0-L-2A, 24 on 0-L-2C and 4 on 0-L-2D are annunciators which alarm if any handswitch on these panels are placed in the "PULL-TO-LOCK" position. These alarms cascade to window 161 -F in the main control room. It was determined that the need for these alarms on PNL 0-L-2 did not exist, therefore, will be removed by this DCN. Whenever a switch is placed in the "PULL-TO-LOCK" position, proper operational clearances and tags are affixed to the switches to identify that condition; therefore, the alarm is not required.

During review of the FSAR for potential impacts by this change, it was found that FSAR Table 11.2-5 (SHEET 4 OF 4) listed the high level alarm setpoint for the CDCT tank. FSAR Table 11.2-5 (SHEET 4 OF 4) will be revised by WBN FSAR Change Package 1440 to change the CDCT high level alarm setpoint.

Safety Evaluation Summary

This change, DCN W-38897-A, does not affect any FSAR evaluations (accident analysis or equipment malfunction failures) previously performed. The accidents and/or malfunctions associated with the Radwaste system is a failure of gas decay tank or associated piping and failure of radwaste components. This change does not affect equipment used in the mitigation of these accident/malfunctions and no new accidents or equipment malfunction failures are created. No radiological consequences are involved due to the fact that release of the CDCT is only permitted after laboratory analysis of the tank contents. This change does not affect the analysis process or increase the radioactive concentration of the CDCT tank and does not involve any credible equipment failure modes since the CDCT level is still monitored using existing level indication and the new alarm setpoint. The CDCT high level alarm is still available to assist Operations in the CDCT filling process and supplements the existing level indication. This change does not cause any additional failure modes and does not interact with equipment important to safety. Technical Specification does not address any operation or safety limit requirements for the CDCT high level alarm and "Motor Locked Out" annunciator portion of the Radwaste System. This change is in compliance with safety classification requirements specified in design basis documents. Therefore, on the basis of the evaluation of effects, it is concluded that the proposed change is acceptable from a nuclear safety standpoint and an unreviewed safety question does not exist.

SA-SE Number *WBPLEE-96-127-1*

Affected Documents

DCN M-38866-A
FSAR Change Pkg. 1441

Document Type

FSAR

Safety Assessment Title

Manual Isolation Valves In Lines To and From the Boron Analyzer and Gross Failed Fuel Detector

Implementation Date:

10/5/96

Description of Change, Test, or Experiments

Previous design changes declared the Gross Failed Fuel Detector (GFFD), the Boron Analyzer and the Turbine Building sump pH monitor would not be used to support Unit 1 operations. However, the associated instruments located on the MCR panels were left in place. For Human Factor reasons, unused devices located on the MCR panels have historically been removed. The previous design changes did not include a-Human Factor Evaluation (HFE). The corrective action required that a HFE be conducted to determine the correct course of action. The HFE concluded that the MCR panel devices should be removed. As implementation of the PER, DCN M-38866-A removes the following devices from the MCR panels: CVCS Letdown Boron Concentration indicator and recorder XI-43-94 and XR-43-94, GFFD flow indicator IFI-43-1002, GFFD count indicator and recorder XI-43-1005 and XR-43-1006 and Turbine Building sump pH monitor alarm annunciator window 169A (window box 15B).

In addition, the indicating light located on the MCR Recording Instrument Board (RIB) for the status of power to the Oil, Oxygen and Acetylene Storage Room water spray control circuit will be removed since it is no longer used. The panel cutouts for the removed devices on MCR panels 1-M-6 and the RIB will be patched and painted to match the overall panel-color. The GFFD devices on panel 1-M-II are mounted in a Westinghouse supplied rack where blank panels will be installed in place of the removed devices. A blank window will be installed in place of the engraved alarm window 169A in window box 15B.

In addition to removal of instruments from the main control board, the design change removes power from remaining instruments and abandons them in place. The power supply cables are determinated at the power source circuit breakers and fuses to deenergize the equipment and isolate them from the supply sources. The containment isolation valves in the line from the Downstream Excess Letdown Heat Exchanger remain as equipment required for Unit 1 and remain fully operational. The manual isolation valves in lines to and from the Boron Analyzer and GFFD are identified as Unit 1/Unit 2 interface points and required to be closed with the handwheel locked closed or removed to isolate the abandoned equipment from system processes.

FSAR Sections 7.7.1.11, Boron Concentration Measurement System; Section 9.3.2, Process Sampling System; and 9.3.5 Failed Fuel Detection System, presently indicate that the equipment is not used and that periodic sampling as described in Section 9.3.2.2 is used to determine boron concentration and delayed neutron activity in the reactor coolant system. The FSAR text in these sections is revised to indicate that the Boron Analyzer and GFFD equipment is "not required" rather than "not used" to convey that the equipment will not be functional and to indicate that the main control room instrumentation has been removed. There is no SAR impact associated with removal of the Turbine Building Sump Ph alarm and removal of the status light on the RIB for the Oil, Oxygen and Acetylene Storage Room water spray control circuit.

Safety Evaluation Summary

This change does not affect any FSAR evaluations (accident analysis or equipment malfunction failures) previously performed. No new accidents or equipment malfunction failures are created. The Technical Specifications are not affected. The SAR presently indicates that the Boron Analyzer and GFFD equipment are not used and no credit is taken for their operation.

Implementation of the GFFD equipment was considered in SSERs 12 and 14, Section 14.2. SSER 14 concluded that the GFFDS performs no safety-related function and is not designed to satisfy any specific safety criteria and that fuel cladding integrity is verified through alternate means.

The changes improve human factors aspects for plant operating personnel, therefore, on the basis of the evaluation of effects, it can be concluded that the proposed change is acceptable from a nuclear safety standpoint and no unreviewed safety question exists.

SA-SE Number *WBPLEE-96-145-0*

Affected Documents	*Document Type*	*Safety Assessment Title*	*Implementation Date:*
DCN W-388871-A	DCN	Replace Rotameter with Differential Pressure Flow Indicators	11/25/97

Description of Change, Test, or Experiments

Change Summary:
This change, DCN W-38871-A, replaces rotameter 0-FI-14-456 with differential pressure flow indicators 0-FI-14-456A and 0-FI-14-456B. The current indicator, O-FI-14-456, is used to indicate condensate demineralizer waste discharge flow to either the Turbine Building sump or cooling tower blowdown. The indicator is used for Offsite Dose Calculation Manual (ODCM) release rate calculations because the waste discharge flow can become radioactive due to a steam generator tube leak. Therefore, 0-FI-14-456 is a compliance instrument in accordance with TI-49. The replacement indicators, O-FI-14-456A and O-FI-14-456B, will become the compliance instruments for the ODCM release rate calculations. The indicators are not safety related in that they are not used to mitigate the consequences of any accident.

0-FI-14-456A and O-FI-14-456B have ranges of 0 - 60 gpm and 0 - 200 gpm, respectively. These two ranges are provided in order to encompass the expected flow rates of between 30 gpm and 150 gpm, and to provide enhanced instrument accuracy at the lower flow rate. The two indicators will be mounted on a local panel in the turbine building near the location of the rotameter, which will be deleted, and the existing sense lines will be reused for differential pressure input to indicators O-FI-14-456A and -456B.

SAR Impact:
This change affects FSAR figure 10.4-36C (drawing 1-47W838-3) by deleting the existing flow indicator 0-FI-14-456 and adding the new flow indicators 0-FI-14-456A and 0-FI-14-456B. This change also affects the Offsite Dose Calculation Manual by deleting and adding the same indicators referenced above in tables 1.1-1 and 2.1-1.

Safety Evaluation Summary

This change provides a different method of measuring condensate polishing demineralizer discharge flow. The method of flow indication has no effect on the occurrence of a design bases accident. This change does not introduce any additional equipment failure modes and is not associated mechanically or electrically with any equipment important to safety. The indicators being added by this change are not safety related in that they are not used to mitigate the consequences of any accident. The indicators do not interact with or affect any equipment used to mitigate the consequences of an accident previously evaluated in the SAR. The replacement of the rotameter with differential pressure units to measure condensate demineralizer discharge flow has no effect on the ability of the steam generator tube bundle to maintain a primary-to-secondary pressure boundary. Also, these flow indicators have no effect on the protective equipment used for the detection and mitigation of a steam generator tube rupture.

This change does not affect any FSAR evaluations (accident analysis or equipment malfunction failures) previously performed. No new accidents or equipment malfunction failures are created. The Technical Specification does not address operation or safety limit requirements for the condensate polishing demineralizer waste discharge flow. This change is in compliance with safety classification requirements as specified in design basis documents. Therefore, on the basis of the evaluation of effects, it can be concluded that the proposed change is acceptable from a nuclear safety standpoint and no unreviewed safety question exists.

SA-SE Number *WBPLEE-96-161-1*

Affected Documents	*Document Type*	*Safety Assessment Title*	*Implementation Date:*
DCN M-38887-A FSAR Change Pkg. 1457	FSAR	Isokinetic Sampling Features	11/29/96

Description of Change, Test, or Experiments

The Service Building Vent Monitor, O-RE-90-132, is required by the FSAR and Design Criteria WB-DC-40-24, to provide for real-time detection of noble gas radioactivity and the capability to collect particulates and iodine for laboratory analysis for the service building ventilation exhaust from the radiochemical laboratory, titration rooms, protection clothing decontamination facility and ventilation room. The design previously implemented also provided isokinetic sampling capability for particulate and iodine monitoring and sampling on adjacent panel O-L-399. The equipment for the isokinetic sampling consisted of a sample flow diverting chamber, sample pump, flow element, flow control valve, and a flow control loop consisting of high and low flow transmitters and indicators, a high select module and a flow totalizer. This design change disables and abandons-in-place the isokinetic sample pump and associated sample flow control instrumentation. Stack flow instrumentation located on panel O-L-399 will be retained. Analytical methods will be used to compensate sample results for particulate loss due to non-isokinetic sample flow.

The sample lines will be capped to isolate sample pump O-PMP-90-320 and flow control valve 0-FCV-90-320. Flow element O-FE-90-320 will be left in place but the tubing to the flow transmitters will be removed and capped. The sample lines will continue to flow to and from Service Building Vent Radiation Monitor O-RE-90-132.

The sample pump will be electrically isolated by disconnecting the 480VAC supply cable at O-RE-90-132. The sample flow instrumentation located on panel O-L-399 will be electrically isolated by disconnecting power supply connections and signal interfaces with the stack flow instrumentation. The instrumentation to be electrically isolated consists of low and high sample flow transmitters O-FT-90-320/2 and 0-FT-90-320/2A; sample high select module O-FM-90-320/3B; high and low sample flow indicators O-FI-90-320/2A and O-FI-90-320/2B; flow controller O-FC-90-320; and flow totalizer O-FQ-3 -320/2. The sample pump control circuit consisting of on-off control switches, status light and starter coil will be disconnected from the 120VAC control circuit.

Front panel instrumentation on 0-L-399, consisting of sample flow indicators, sample flow totalizer, and sample pump control switches and status light, will be removed in accordance with human factors design practice.

FSAR Section 11.4.2.2.4, Ventilation Monitors and Containment Atmosphere Monitors, Service Building Ventilation Monitor, states that real time monitoring of particulate and iodine activity is not required. The section is being revised to remove a portion of the text that describes the sample nozzle design and implies the use of isokinetic sampling ("representative sampling").

Safety Evaluation Summary

This change does not affect any FSAR evaluations (accident analysis or equipment malfunction failures) previously performed. No new accidents or equipment malfunction failures are created. The Technical Specifications are not affected. The SAR presently indicates that only gaseous monitoring which does not need isokinetic sampling to achieve a viable sample is required for the Service Building Ventilation Monitor (ODCM Table 1.1-2).

Implementation of the Radiation Monitoring System including the Service Building Ventilation Monitor was considered in SSERs 16 and 20, Section 11.5. The SSERs concluded that process and effluent radiological monitoring system as described in the SAR for Watts Bar Unit 1 complies with the acceptance criteria of SRP 11.5 and is acceptable.

The change does not eliminate or alter required equipment or functions, therefore, on the basis of the evaluation of effects, it can be concluded that the proposed change is acceptable from a nuclear safety standpoint and no unreviewed safety question exists.

SA-SE Number *WBPLEE-96-162-0*

Affected Documents	*Document Type*	*Safety Assessment Title*	*Implementation Date:*
DCN W-39066-A	DCN	Setpoints for Auxiliary Building floor and equipment drain sump pumps A and B level control switches	2/25/97

Description of Change, Test, or Experiments

This change, DCN W-39066-A, revises the level switch setpoints used to monitor and control the Auxiliary Building floor and equipment drain sump pumps A and B. The subject level switches perform the following functions; 1). 0-LS-077-0133M, O-LS-077-133AB, O-LS-077-133BA, and O-LS-077-133BB (pump "on" signal); 2). 0-LS-077-0133DA and O-LS-077-0133DB (pump "stop" signal); 3). 0-LS-077-0133E (high level alarm); 4) 0-LS-077-0133F (low level alarm). All of the above level switch setpoint elevations are increased by a value of 6-inches.

This change is in response to a corrective action step for WBPER960338. The PER documented a condition where air flow was detected from the sump to the floor drain openings caused by slight differences in pressure between Auxiliary Building rooms/areas. This change will maintain the sump minimum water level above the sump drain line connection elevation, thus, creating a water seal effect. This water seal will prevent air flow from the sump to the floor drain openings.

Auxiliary Building floor and equipment drain sump and associated pumps A and B are part of the liquid radwaste processing system (LRPS), which is designed to receive, segregate, process, and discharge liquid wastes. The Auxiliary Building floor and equipment drain sump collects liquids from floor drains and equipment leaks. This system is discussed in FSAR Sections 9.33 (Equipment and Floor Drainage System) and 11.2 (Liquid Radwaste Systems), and setpoints are shown in Figure 9.3-8 (Mechanical Flow Diagram, Floor and Equipment Drains).

This change has no special requirements and can be worked in any mode when in accordance with and allowed by normal plant procedures. The liquid radwaste processing system is designed to operate during all normal modes of plant operation.

Safety Evaluation Summary

The switches impacted by this change are not safety-related and are not used in T/S compliance monitoring. Therefore, they are not used in any accident analysis. This changes does not affect the operation of the Auxiliary Building passive sump which is used to contain any postulated line break. Changes to the setpoints for this nonsafety-related system do not affect radwaste system processing, but only the level in this sump at which pumping from the sump to a radwaste processing begins or stops. Therefore, existing failure modes are not changed.

This change does not affect any FSAR evaluations (accident analysis or equipment malfunction failures) previously performed. No new accidents or equipment malfunction failure are created. This change is in compliance with safety classification requirements as specified in design basis documents. The Safety Evaluation Report (SER) and Supplementals have been reviewed and no impacts were found. Therefore, on the basis of the evaluation of effects, it is concluded that the proposed change is acceptable from a nuclear safety standpoint and no unreviewed safety question exist.

SA-SE Number *WBPLEE-96-170-1*

Affected Documents

TS Bases Change Pkg. 96-020
TSB SR 3.4.12.7

TS Bases Revision 7

Document Type

Tech Spec Bases

Safety Assessment Title

Change clarifies the requirement to perform a Channel Operational Test (COT) of the cold overpressure mitigation System (COMS) instrumentation that controls automatic RCS PORV actuation when RCS temperature is < 350 degrees F (Mode 4).

Implementation Date:

Description of Change, Test, or Experiments

Change Summary:
This Technical Specification Bases change package revises section SR 3.4.12.7 of the Bases in order to clarify the requirement to perform a Channel Operational Test (COT) of the Cold Overpressure Mitigation System (COMS) instrumentation that controls automatic RCS PORV actuation when RCS temperature is < 350°F (Mode 4). The wording of the Technical Specification Bases currently implies a requirement to perform a COT within 12 hours after decreasing RCS temperature to < 350°F even when the COT has been performed within the 31 day Technical Specification requirement of SR 3.4.12.7. The Technical Specification allows 12 hours to satisfy the COT surveillance requirement after entering mode 4 assuming the COT had not been performed in the 31 days prior to entering mode 4. The wording provided in the Bases for SR 3.4.12.7 does not correctly interpret the intent of the requirement. Therefore, the Bases is being revised to clarify a COT may be performed for the COMS instrumentation in the 31 days prior to entering mode 4 to be sufficient for meeting surveillance requirement SR 3.4.12.7.

SAR Impact:
This change affects the Technical Specification Bases, SR 3.4.12.7. No other SAR impacts were identified.

Safety Evaluation Summary

The Cold Overpressure Mitigation System (COMS) is provided as a backup to the operator in order to minimize the potential for an RCS overpressurization. This pressure control system includes manually armed semi-automatic actuation logic for the two pressurizer Power Operated Relief Valves (PORVs). The function of this actuation logic is to continuously monitor RCS temperature and pressure conditions, with the actuation logic only unblocked when plant operation is at a temperature below the arming setpoint. The actuation logic and relief valves are redundant and independent. This system provides the capability for RCS inventory letdown, thereby maintaining RCS pressure within allowable limits. The COMS does not perform a protective function, but rather provides automatic pressure control at low temperatures as a backup to the operator. The Technical Specification Bases change does not alter the COMS instrumentation, the programmed setpoint, or the method of performing the Channel Operational Test (COT). The Bases change does not change the frequency required by the Technical Specifications for performing the COT for COMS. The Bases change only allows the plant to adhere to the intent of the Technical Specifications without performing an additional COT after entering mode 4 during a cooldown when the Technical Specification frequency requirement for performing the COT has been met. This change allows the COMS instrumentation to be tested for operability prior to entering mode 4 during cooldown, which provides assurance of meeting the limits established in the RCS Pressure and Temperature Limits Report. No new accidents or equipment malfunction failures are created. This change does not affect any FSAR evaluations (accident analysis or equipment malfunction failures) previously performed. This change is in compliance with safety classification requirements as specified in design basis documents. All instrumentation being tested is located in a mild environment in the auxiliary instrument room and is not affected by RCS temperature excursions. Therefore, on the basis of the evaluation of effects, it can be concluded that the proposed change is acceptable from a nuclear safety standpoint and no unreviewed safety question exists.

SA-SE Number *WBPLEE-96-186-0*

Affected Documents

DCN S-39014-A
FSAR Change Pkg. 1429
SOI-100.01

Document Type

FSAR

Safety Assessment Title

Revision of Telephone and Radio Systems

Implementation Date:

5/5/97

Description of Change, Test, or Experiments

Detailed Description: DCN S-39014-A revises N3-2504003 "Automatic, Manual, and Public Telephone System" and N3-2534003 "VHF Radio and MW Radio System", and FSAR sections to reflect actual plant communication configuration.

N3-250-4003 "Automatic, Manual, and Public Telephone System"

1. Removed reference to Bell's PAX & Dimension 2000 telephone systems and made reference to the current Telephone System (TSS) which is the Fujitsu telephone system installed by the Harris company.
2. Revised organizations names throughout (Customer Group became Transmission Power Supply & Nuclear Security Services became Nuclear Security).
3. Deleted references to "power line carrier".
4. Deleted term "dedicated" as it relates to the Emergency Notification System.
5. Revised the description of the power supply for the telephone system.
6. Clarified the location of the "attendant console".
7. Deleted reference to a "manual" telephone system.
8. Identified that Node 2 is part of the Automatic Telephone System.
9. Clarified the location of TSS alarms.
10. Clarified where telephones can be disabled during an emergency.
11. Clarified that programming is not done at the attendant console.
12. Clarified that the manual telephone is not used to answer emergency calls for fire.
13. Deleted the statement that the DOD trunks would automatically be routed if the TSS system completely failed.

N3-253-4003 "VHF Radio and MW Radio System"

1. Removed reference to DS-E11.17.1 (was inactivated).
2. Revised MW Radio System-Circuit 8630 description.
3. Deleted MW Radio System-Circuit 8780.
4. Clarified the Emergency Radio Communications System.
5. Added the receiving frequency for F3 Repeater.
6. Corrected the F4 Repeater transmit frequency.
7. Revised organizations names throughout (Customer Group became Transmission Power Supply & Nuclear Security Services became Nuclear Security).
8. Clarified the purpose of the Control interface Units.
9. Corrected the transmit frequency for the high band VHF repeater.
10. The description of third branch of the Emergency Preparedness Radio Communications System antenna was added.
11. Revised the description of the base station control of the NS Sheriff's Radio.
12. Revised the model numbers of the NS Sheriff's Radio remote control units.

FSAR Section 2.4.14.9.6 "Communications Reliability"

Safety Evaluation Summary

This change does not affect any SAR accident analysis or equipment malfunction failures previously performed. The proposed changes only affect the plant communications system which is not safety-related equipment. The change will not require any modifications of any equipment important to safety. No new failure modes are introduced and equipment reliability is not degraded by this change. Therefore, the changes will not increase the probability of an accident or malfunction of the plant communications systems or any equipment important to safety previously evaluated in the SAR. The proposed changes have no affect on any accident with dose consequences. Therefore, these changes do not increase the radiological consequences of any accidents previously evaluated in the SAR.

A portion of the changes described are related to communications systems which are used to provide early notification to NRC and the general public. The changes described does not reduce the ability for WBN to communicate effectively onsite and offsite should a dose accident occur. This change is in compliance with safety requirements as specified in design basis documents. Therefore, on the basis of the evaluation of effects, it is concluded that the proposed change is acceptable from a nuclear safety standpoint and no unreviewed safety question exist.

Revise to reflect the communications methods between projects within the TVA power system.

FSAR Section 2.4.14.10 "Basis for Flood Protection Plan in Seismic-Caused Dam Failures"

Revise to reflect Watts Bar Nuclear does not communicate with the TVA power system using a power line carrier.

FSAR Section 8.3.2.1.2 Non-Safety Related DC Power Systems"

Revise to remove reference to PAX and power line carrier equipment.

FSAR Section 9.5.2 "Plant Communications Systems" requires the following changes:

1. Deleted references to "power line carrier".
2. Revised the description of the power supply for the telephone system.
3. Deleted the location of the Health Physics sound powered telephone on Unit 2
4. Revised the description for the Closed Circuit Television System.
5. Revised the Evacuation" alarm to the Assembly and accountability" alarm.
6. Deleted statement that any TSS telephone could be used to initiate the fire and medical alarm.
7. Removed reference to the "Montlake Repeater Station
8. Revised the purpose of the Health Physics Network (HPN) and Emergency Notification System (ENS). Revised the location of the telephones which provide access to the HPN.
9. Revised the location where the commercial telephone lines are terminated.
10. Identified that the nuclear security and sheriff's radio system has components (cables) in the communications room.
11. Revised FSAR Figure 9.5-19 to address the above items.
12. Revised organizations names throughout (Customer Group became Transmission Power Supply & Nuclear Security Services became Nuclear Security).
13. Removed FSAR Figures 9.5-16 & -17.
14. Removed location of code call and paging equipment.
15. Removed the list of radio & microwave equipment.

FSAR Impact
The above are only changes related to the configuration of Watts Bar communications systems and do not affect plant safety.

Safety Evaluation Report (SER) June 1982 (including supplements 1 through 20) is impacted. The SER reflects the current sections prior to the proposed changes in FSAR change package 1429.

SA-SE Number *WBPLEE-96-191-0*

Affected Documents	*Document Type*	*Safety Assessment Title*	*Implementation Date:*
DCN M-39082-A	DCN	Time Constants for the Auctioneered Reactor Coolant Average Temperature Lead/Lag Module	5/6/97

Description of Change, Test, or Experiments

DCN M-39082-A modifies the time constants for the auctioneered reactor coolant average temperature (Tavg) lead/lag module 1 TY-412P, which provides input to the reactor control system to produce rod speed and direction signals when the rod control system is in the automatic mode. The purpose of this change is to minimize rod stepping due to fluctuations in primary coolant average temperature by reducing the sensitivity of the control system to these fluctuations.

The reactor control feature of the Control Rod Drive System consists of an automatic system designed to maintain a programmed average temperature in the Reactor Coolant System by regulating the core reactivity. During steady-state Operation, the control system maintains Tavg within +3.5°F of the programmed reference temperature (Tref). The control system is designed to operate in automatic mode above 15% reactor thermal power. The reactor control signal consists of an error signal used to direct rod speed and position to automatically control reactor power. The two channels used to generate the total error signal are (1) the deviation of the actual auctioneered (highest loop) Tavg from the programmed temperature Tref and (2) the mismatch between turbine load and nuclear power as represented, respectively, by turbine impulse pressure and auctioneered NIS power signal. Signal compensation is provided on both channels to ensure proper control system response to plant transients. The power mismatch circuits provide fast response to a load change transient by anticipatory compensation, while the temperature mismatch circuits primarily provide steady-state control. To minimize rod stepping, the lead and lag time constants will be modified. The lead time constant will be decreased to decrease the rate compensation from temperature spikes and one of the lag time constants will be increased to filter out temperature spikes.

The safety function of the control rod system is to provide the availability of sufficient rod worth which, when inserted will maintain the reactor core in a subcritical condition at any point during the fuel cycle. The safety function of the control rods is accomplished independently of the control system by interruption of power to the rods. Although the automatic rod control system is a non-safety grade system, its operation is modeled in the safety analyses when it makes the analysis results more limiting. Those events which model normal, automatic operation of the rod control system were evaluated. Other events are not impacted by this change since they conservatively assume a rod control system failure (as either an initiating event or a result of an initiating event) or they assume the unit is operating in manual rod control. The results of these evaluations demonstrate that, while the proposed changes could make the event more limiting with respect to DNB, the DNB design basis continues to be met. The time constants will be set conservatively with respect to the values used in the analyses. A lead time constant greater than the analysis value or a lag time constant less than the analysis value are conservative since either will increase the sensitivity of the system to temperature changes and would make the event results less limiting. The descriptions of the rod control system and the affected event analyses in the SAR remain valid.

Safety Evaluation Summary

The change described above does not affect any SAR evaluations (accident analysis or equipment malfunction failures) previously performed. The decreased sensitivity of the control system will slow the response of the system to transients and, in some cases, results in making the event more limiting with respect to DNB. The changes are within the design limitations of the hardware and will not result in increased instability of the system or decreased equipment reliability. The decreased sensitivity resulting from the time constant changes will minimize rod stepping, thereby contributing to more stable reactor operation and improved reactivity control during normal operation. An evaluation of affected events performed by Westinghouse concluded that all acceptance criteria continue to be met and that the results and conclusions documented in the SAR remain valid. No new accidents or equipment malfunction failures are created and no Technical Specifications are affected. The change affects only a non-safety grade system which has no accident mitigation function. Therefore, on the basis of the evaluation of effects, it can be concluded that the proposed change is acceptable from a nuclear safety standpoint and no unreviewed safety question exists.

SA-SE Number *WBPLEE-97-006-2*

Affected Documents	*Document Type*	*Safety Assessment Title*	*Implementation Date:*
DCN M-39207-D FSAR Change Pkg. 1477	DCN	Conduit and Grounding Details for Stage 2	10/13/97

Description of Change, Test, or Experiments

DCN M-39207-D implements the necessary changes that will allow the unit 1 generator output (through generator breakers PCB 5044 and 5088) to be tied to two 500 KV switchyard buses (1 and 2) or tied to each one separately. The change also allows Unit 2 to be backfed from bus 1 through Unit 2 main transformer while Unit 1 generator is tied to bus 2. The different 500 KV switchyard configurations can be achieved by switching 2 new Motor Operated Disconnects (MOD). One of these MODs (6117) is used to isolate the high side of the Unit 1 main transformer from the Unit 2 generator breaker (PCB .5044). The other MOD (6127) is used to isolate the Unit 2 main transformer from the 500 KV switchyard (bus 1). The differential relaying circuits have been modified to only allow the Unit 1 generator to be tripped by bus 1 when it is connected to the Unit 1 generator output. The breaker failure relaying circuits have been modified to allow bus 1 or 2 to be isolated from the Unit 1 generator connection when a Line breaker failure occurs. This is accomplished by tripping all line breakers tied to the faulted bus and the generator breaker supplying the faulted bus. This has the advantage of allowing the unit to stay on line for a fault on a given bus. If either generator breaker (PCB 5044 or 5088) experiences a breaker failure the unit will be tripped and the switchyard isolated. PCB 5064 is a spare breaker that can be used to replace any breaker in the 500 KV switchyard. Its controls and protective circuits have been modified to allow it to function exactly as the breaker it replaces.

This change results in significant savings in transmission and transformer losses.

FSAR Section 8.1.1, Utility Grid and Interconnections, Paragraph 2 presently discusses that both nuclear units are connected into TVA's 500 KV transmission system. One unit is connected with three and the other with two 500 KV lines which are integral parts of the 500 KV transmission system. The proposed FSAR change will indicate that the plant is connected into a strong 500 KV transmission system and that the 500 KV switchyard has two bus section, one with two 500 KV lines and the other three 500 KV lines. Unit 1 is connected to each bus section through generator breakers that tie the two bus sections together electrically.

FSAR Section 8.2.1 , Paragraph 4 first sentence will be changed to indicate there are more than one main generator breaker by changing generator breaker to generator breaker 1 and/or 2 and the word breaker to breakers.

FSAR Section 8.2.2, Paragraph 3 will be revised to indicate that the transient stability studies show that for a stuck breaker on one bus the Unit remains connected to the transmission system through one breaker, the unfaulted bus, and its associated transmission lines. For all cases studied , WBN Unit 1 power swings are damped, and the unit maintains synchronism with the system. Also, the 161 KV offsite power supply voltage at Watts Bar Hydro Recovered to sufficient levels after the faulted bus was cleared.

FSAR Figure 8.2-2 will be revised to reflect the new configuration.

FSAR Figure 8.1.-1 is also being updated to reflect the latest information.

Safety Evaluation Summary

This change allows the Unit 1 generator to be connected to either 500 KV switchyard bus 1 and/or 2 and does not affect any protection features required for the initiation of safety injection or other safeguards and does not alter the design, function, or performance of any safeguards equipment. This feature presently exists for Unit 1 generator breaker and bus 2 differential and breaker failure relaying and will be incorporated for Unit 2 generator breaker and bus 1 differential and breaker failure relaying. Thus, this change does not affect any FSAR evaluations (accident analysis or equipment malfunction failures) previously performed. No new accidents or equipment malfunction failures are created. Thus, for some 500 KV switchyard events such as a fault and a stuck breaker, one bus can be isolated while the main generator remains connected to the unfaulted bus. Before the modification the probability of a unit trip due to a 500 KV switchyard event was one trip every 2.9 years and after the modification the probability is one trip every 19.4 years. The change does not alter the design, function, or performance of any safeguards equipment, and will not impact any operator actions which may be required after an event. Therefore, it is concluded that the radiological consequences analyzed in the FSAR remain unchanged.

The impact of the change was evaluated against the design and licensing basis and Technical Specification Bases requirements and was found to be acceptable and will not degrade any systems or components required for accident mitigation. It is therefore concluded that no margins of safety will be reduced by the proposed activity. Therefore, on the basis of the evaluation of effects, it can be concluded that the proposed change is Acceptable from a nuclear safety standpoint and no unreviewed safety question exists.

FSAR Section 10.2.4 will be revised to reflect that generator 1 and/or 2 can trip the turbine.

The FSAR discussions in Sections 7.2.1.1.2.4 and 15.3.4.1 concerning the worst case 500 KV grid maximum frequency decay rate of less than 5 Hz/sec was re-evaluated and is still valid for all possible 500 KV switchyard configurations following the proposed change.

It should be noted that SER discussions are impacted by the proposed change. The June 1984 SER, section 10.2.1 will be affected in that after the proposed change generator breaker 1 and/or 2 and bus 1 differential and breaker failure relaying can shutdown the turbine. SER Supplement 13 section 8.2.2.3 will be impacted in that after the proposed change the generator breaker involvement will be generator breaker 1 and/or 2 instead of generator breaker.

SA-SE Number *WBPLEE-97-058-1*

Affected Documents	*Document Type*	*Safety Assessment Title*	*Implementation Date:*
DCN W-39242-A	DCN	Replacement of Vibration Monitoring Portion of the Main Turbine Supervisory Instrumentation	10/13/97

Description of Change, Test, or Experiments

This DCN (M-39242-A) replaces the vibration monitoring instrumentation of the Main Turbine Supervisory Instrumentation (TSI) system in unit 1. The TSI system is a part of the Turbogenerator Control and Protection System (TGCPS) and is used to sense, indicate, record and alarm a number of measured variables on the steam turbine. The information provided by the TSI is used by the operator to make judgment as to whether the unit is operating in a stable condition or needs to be operated at a reduced load or shut down for inspection and repair. The following variables are measured by the TSI system. All existed for the-previous instrumentation except coupling vibration, which is added by this modification.

a. Governor Valve Position
b. Rotor Speed
c. Shaft Eccentricity
d Case Expansion
e. Differential Expansion
f. Rotor Position (Thrust Position)
g. Shaft Vibration
h. Coupling Vibration

The information gathered by the TSI is displayed in the main control room via a CRT display indicating the above parameters. If an abnormal condition occurs, the appropriate annunciator will sound to alert the operator to mitigate corrective action. There are hi and hi-hi alarms for Differential Expansion, Rotor Position, Shaft Vibration, and Coupling Vibration. There is a high alarm for Shaft Eccentricity. The alarm features and corresponding setpoints are not changed by this modification, except that the added parameter coupling vibration is paralleled with the other shaft vibration alarms. The operator is expected to take manual action to trip the turbine on hi-hi vibration since the auto trip on high vibration has been and remains blocked by handswitch 1-HS-47-120.

(The turbine trip system is also equipped with solenoid-operated trip devices, which provide a means to initiate direct tripping of the turbine upon receipt of appropriate electrical signals. Also, when a turbine trip is initiated, the extraction system nonreturn valves are tripped to close by means of a pilot dump valve connected to the turbine trip system.)

The turbine is tripped manually on detection of high temperature, high back pressure in any pressure zone. high journal or thrust bearing metal temperatures, high bearing oil discharge temperature, excessive shaft vibration, or high differential expansion.

All turbogenerator protective trips that will automatically trip the turbine due to turbine (mechanical) abnormalities are tabulated below.

Automatic Turbine Trips Due To Turbine (Mechanical) Abnormalities
1. Low Bearing Oil Pressure Trip
2. Low Vacuum Trip
3. High Thrust Bearing Trip

Safety Evaluation Summary

This change upgrades existing plant equipment The failure modes of the replacement equipment do not differ from the equipment being replaced. The new Bentley Nevada equipment monitors for failure modes and any problems detected are presented to operations on the ERFDS system. The TSI system is not credited with mitigating any FSAR Chapter 15 events. The TSI system interfaces only with control board indicators and the annunciator system, neither of which have a primary safety function. The potential for inadvertent manual trip of the turbine based on inaccurate alarms and indications is decreased due to the higher reliability of the equipment and its self-diagnostic capabilities. Though the loss of ERFDS would result in the loss of the main control board LCD display, the information is available at the Data Manager 2000 in the compute-room and on individual indicators on the turbine deck. Speed and governor valve position would still be available on the main control board via the FHC system and the annunciator alarms would not be affects by the loss of ERFDS. Proper separation/isolation of cable routing and equipment is maintained by the DCN and appropriate plant installation procedures. This includes drawing 45WI640 for control boards. The independence of safety related equipment is not challenged. Civil calculations have been performed to verify that, when installed per the DCN. the equipment will retain it's position in a seismic event. Purchase requisitions 97C-108 and 97C-158 requires that EMI/RFI radiated emissions will be within the requirements of DS-E18.14.01 and not adversely affect the operation of surrounding equipment. This change will not compromise the ability of plant safety-related equipment to perform its intended function.

4. Low Differential Water Pressure Across Generator Stator Coils Trip (Alarm only below 15% power)
5. High Stator Coil Outlet Water Temperature Trip (Alarm only below 1~% porter)
6. Low Lube Oil Tank Level
7. Low Auto Stop Oil Pressure Top
8. 111% Rated Speed Electrical Overspeed Trip
9. 110% Rated Speed Mechanical Overspeed Trip
10. EHC dc Power Failure Trip
11. Loss of Both Main Feedwater Turbines Trip
12. Steam Generator High-High Level Trip

There are no WBN design basis events for which the TGCPS is required to operate. The major plant safety concern for the TGCPS is the prevention of generation of turbine missiles due to a turbine overspeed condition (uncontrolled run away of the turbine). As referenced above, there is a mechanical overspend trip mechanism to trip the turbine at 110 % of speed. There is also an electrical overspeed trip at 111% of rated speed.

The present system is outdated with no replacement parts available. Westinghouse now supplies Bentley Nevada Corporation TSI as standard equipment in all new turbo-generators and for all TSI retrofits.

FSAR Figure 10.2.4 (1-47W610 17-3) is revised due to this modification.

SA-SE Number *WBPLEE-97-065-1*

Affected Documents	*Document Type*	*Safety Assessment Title*	*Implementation Date:*
DCN M-39251-A	DCN	On-Delay SI Reset Timing Relay in the SSPS	9/24/97

Description of Change, Test, or Experiments

DCN M39251-A replaces the on-delay SI reset timing relays in the SSPS with relays which have both on-delay and off-delay. This change is being made in response to Westinghouse Infogram 96-004 which described a potential failure of the SI reset function due to "relay race." SI actuation results in energizing multiple latching slave relays in the SSPS, one of which, K602, energizes the SI reset relay TD1. TD1 is a timing relay whose contacts remain open until the relay times out. TD1 contacts are interlocked with the SI reset switches in the main control room (MCR) to prevent actuating the unlatch coils of the SI slave relays and thereby prevent establishing SI block to the logic circuits. After TD1 times out (90 seconds), its contacts close, enabling the manual SI reset function from the MCR. The reactor trip breakers must also be open (P4 signal) to compete the SI block. The SI reset switch energizes the unlatch coils of the SI relays and combines with P4 to generate the SI block. The SI block removes the SI actuation signal from the logic circuits which operate the SI relays. Once the unlatch coils are energized, the K602 latch coil drops out and will not re-energize because the SI actuation signal is no longer present. The K602 contacts then open and cause TD1 to de-energize which then disables the manual SI reset circuit. If K602 drops out before the other SI relays are unlatched, they will remain latched (actuated). If this were to occur, further operation of the SI reset switch would not unlatch these relays. This problem can be averted by delaying the de-energizing of TD1 until the slave relays have had time to completely unlatch.

DCN M-3S251-A replaces TD1 in each of the SSPS output cabinets (Train A and Train B) with a timing relay which has both time delay on and time delay off functions. The on-delay will remain set at 90 seconds and the off delay will be set at 5 seconds. The addition of the off-delay will not prevent actuation of any safeguards loads but will permit the SI reset function to be accomplished without risk of some of the slave relays remaining in the energized state. The replacement relay will be qualified for Class 1E service. No wiring changes will be required and the SI reset operator Interface wit not be affected.

The SI reset feature is provided to allow operator control of equipment after the initial phase of injection (e.g., to secure safety loads not required to mitigate the current event or to re-energize pressurizer heater control). The time delay of 90 seconds allows sufficient time for ESF loads automatically actuated by SI to reach their accident mitigation condition. The time-delay relay provides capability, after the delay, to reset the SI slave relays and block the SI signal; the actuation of ESF equipment is independent of the relay. The addition of the off delay merely ensures that the reset function will operate as intended and will have no effect on operator actions that may be required after an SI.

The SI signal reset function is mentioned in various sections of the FSAR (e.g., 7.3.2.2.6) and shown on FSAR Figure 7.3-3 Sheet 3. Although the 90 second time delay value for the signal reset is not given in the SAR text, the drawing on which the figure is based, 1-47W611-63-1, does show the time delay setting. The drawing and figure will be revised in accordance with plant procedures to show the addition of the off-delay to this function.

Safety Evaluation Summary

The addition of the off-delay to the SI reset function will not affect initiation of safety injection or other safeguards functions nor will it impact any operator actions which may be required after an event. The change does not affect any FSAR evaluations (accident analysis or equipment malfunction failures) previously performed and no new accidents or equipment failure modes are created. The Technical Specifications are not affected. This change is in compliance with safety classification requirements as specified in design basis documents. The change will enhance protection system reliability since it will ensure that the SI reset function will operate as intended. Therefore, on the basis of the evaluation of effects, it can be concluded that the proposed change is acceptable from a nuclear safety standpoint and no unreviewed safety question exists.

SA-SE Number *WBPLEE-97-067-0*

Affected Documents	*Document Type*	*Safety Assessment Title*	*Implementation Date:*
DCN W-39386-A	DCN	RCS Loop Tavg/Auctioneered Tavg Deviation Alarm Setpoint and Reset Value Revised	5/23/97

Description of Change, Test, or Experiments

DCN W-39386-A revises the RCS Loop Tavg/Auctioneered Tavg deviation alarm setpoint from +2°F to +3°F and the reset value from 1% to 0.5°/0. This alarm is annunciated in the main control room (MCR) on window 5A-94B and is actuated when the difference between any loop Tavg and the auctioneered high Tavg of the four RCS loops exceeds the setpoint. The present setpoint is 2°F and has a deadband of 1% of span, which is equivalent to 1°F. At full power, the two loops with the highest and lowest average temperatures have been operating with a nominal difference in loop Tavg of up to 1.8°F. With this nominal difference, RCS loop temperature fluctuations of even small amounts can actuate the alarm. These temperature fluctuations have been observed in other plants and are the result of hot leg streaming or upper plenum anomaly. Since the deviation does not drop to less than the 1 °F reset value, the alarm will not reset and the annunciator window remains lit. The alarms are actuated by bistables in the plant process control instrument racks.

The Tavg deviation alarm provides an alert that the plant may be operating outside normal steady state conditions. It may also be indicative of other abnormal conditions such as failure of hot or cold leg instrumentation, steam flow/feed flow mismatch, or reactor coolant pump trip. Additional alarms are provided in the MCR for all of these conditions. The current 2°F alarm value was selected to distinguish between normal loop deviations and abnormal operating conditions based on engineering evaluation of historical operating data. Due to modifications such as more aggressive core designs, the normal loop differences for Tavg in some plants can approach the 2°F value. Increasing the alarm setpoint to a value of +3°F and reducing the reset to 0.5% (0.5°F) will eliminate the nuisance alarm condition while preserving the intended function of notifying the operator of plant operation outside steady state conditions.

These alarms are mentioned in Section 7.2 of the FSAR but no setpoint is specified. However, in paragraph 4.4.3.4 of Supplement 8 of the Watts Bar Safety Evaluation Report, a description of the Tavg deviation alarm was provided and included the setpoint of 2°F. This evaluation was based on TVA submittals for the RTD Bypass Elimination project, including a July 9, 1991, response to an NRC request for additional information (RAI) dated January 8, 1991, which specifically requested information concerning AT and Tavg loop deviation alarms and setpoints.

Safety Evaluation Summary

The loop Tavg deviations are not used as input to any protection functions and there are no associated Technical Specifications or Technical Requirements. There are no control functions associated with these loop differences and, thus, the alarm setpoint change will have no impact on control systems which use Tavg as an input (e.g., rod control system). The change does not involve any new or different type of equipment or hardware modifications and, therefore, no additional or different failure modes will be created. Revising the setpoint provides a benefit in that it will reduce nuisance alarms which can detract operator attention from more important tasks.

The change satisfies the intent of the alarm as specified in the design basis and does not affect the input assumptions to any safety analyses. The safety analyses do not model or take any credit for operator action associated with this alarm and do not explicitly model loop to loop variations in Tavg. They generally assume the use of a nominal design Tavg that is consistent with the Tavg program and adjusted for uncertainties. Thus, the change does not affect any SAR evaluations (accident analysis or equipment malfunction failures) previously performed and no new accidents or equipment malfunction failures are created. Therefore, on the basis of the evaluation of effects, it can be concluded that the proposed change is acceptable from a nuclear safety standpoint and no unreviewed safety question exists.

SA-SE Number *WBPLEE-97-069-0*

Affected Documents

TRM Change Pkg. 95-093
TSR 3.1.7.1

TRM Revision 8

Document Type

Technical Requirements Manual

Safety Assessment Title

Additional Surveillance Testing of Group Demand Step Counters

Implementation Date:

9/22/97

Description of Change, Test, or Experiments

This change revises the Technical Requirements Manual (TRM) (Change Package No. 95-093) to clarify the frequency statement related to Technical Surveillance Requirement (TSR) No. TSR 3.1.7.1. Also, this safety assessment covers the associated revision to the TSR implementing Technical Requirements Instruction No. 1-TRI-85-1.

TSR 3.1.7.1 requires the determination that each control rod group demand step counter is OPERABLE. This requirement is applicable in modes 3, 4, or 5, when the reactor trip breakers (RTBs) are closed, and the associated rod group is not fully inserted. This surveillance is accomplished by movement of the associated shutdown and control rods 10 steps in any one direction and a verification that the step counters for each group (within each bank) are within 2 steps of each other. The associated surveillance frequency statement is as follows: Within 4 hours after closing the reactor trip breakers - AND - 31 days thereafter". This TRM change package will clarify the 4 hour surveillance requirement to be applicable only if the TSR is outside the 31 day frequency. The revised frequency statement for TSR 3.1.7.1 will be as follows: Within 4 hours after closing the reactor trip breakers if not completed within the previous 31 days - AND - 31 days thereafter."

This TRM change is necessary to clarify the intent of the operability requirement specified in TR 3.1.7. When the group demand step counters are required to be operable, the TRM Bases states that 4-hours is provided to perform the first surveillance after closing the RTBs. Thereafter, exercising the rods at a frequency of 31 days provides sufficient confidence that the counter continues to be operable. The 4-hour requirement was included in the TRM to allow sufficient time to perform the operability surveillance. In modes 3, 4, and, 5 the RTBs are open and closed several times to accommodate testing required for plant startup. As long as the TSR is within the frequency of 31 days, and no other maintenance or testing activity has otherwise jeopardized the operability of the counters, then the 4-hour surveillance following a subsequent closure of the RTBs would not be required.

Safety Evaluation Summary

This TRM change does not affect any SAR evaluations (accident analysis or equipment malfunction failures) previously performed. All design basis events which require control rod insertion are detected and mitigated by the Reactor Protection System which is not impacted by this change. Technical Specification is not affected. No new accidents or equipment malfunction failures are created. Specifically, the disruption of the MG sets generated supply power to the Control Rod Drive System due to RTB operation, does not significantly increase the probability of failure of the group demand step counters. The Logic Cabinet, which controls counter operation, uses an alternate power supply from an independent power source. The Logic Cabinet is designed with an auctioneering circuit to automatically select the best available power supply. Thus, no new equipment malfunction failures are created. This change is in compliance with safety requirements as specified in design basis documents. Therefore, on the basis of the evaluation of effects, it is concluded that the proposed change is acceptable from a nuclear safety standpoint and no unreviewed safety question exist.

SA-SE Number *WBPLEE-97-072-0*

Affected Documents	*Document Type*	*Safety Assessment Title*	*Implementation Date:*
DCN W-39345-A	DCN	Waste Gas Compressor Unloading Pressure Switch Setpoints Changed	7/16/97

Description of Change, Test, or Experiments

DCN W-39345-A revises the setpoint of pressure switches (0-PS-77-92 in stage 1 & 0-PS-77-106 in stage 2) to 105 psig. These switches are used to unload the Waste Gas Compressors 0-COMP-77-91 & 0-COMP-77-105 when the pressure at the compressor seal inlet reaches 90 psig. The pressure developed by the compressor is used to pressurize gas decay tanks. When the "in service" gas decay tank is pressurized to 100 psig, flow is automatically switched to the "standby" gas decay tank. The current output of 90 psig causes the compressors to unload which prevents the pressure from reaching 100 psig and causes the compressors to run excessively. The new setpoint of 105 psig will allow the compressors to pressurize the-gas decay tanks and prevent the compressors from excessive run times.

Radioactive waste gases vent directly to the vent header which leads to the suction of the waste gas compressors. The compressors can be operated either in manual or automatic. From the compressor, the compressed gas flows into one of nine gas decay tanks. Control arrangements for the gas decay tank inlet header allow the operator to place one gas decay tank in service and select another as a standby. When the "in service" tank reaches a pressure of 100 psig, flow is automatically switched to the "standby" tank. The gas unloading valve is designed to open at a preselected value of between 60 and 120 psig. This unload valve is actuated in response to the unloader pressure switch which senses pressure in the compressor seal inlet area. The waste gas compressors are not safety related and are not required to function during or after a design basis event.

This DCN can be implemented in any plant mode and in any stage sequence.

The information presented in the SAR for the waste gas compressors is not impacted by this modification.

Safety Evaluation Summary

The changes described above for the waste gas compressors do not affect any SAR evaluations (accident analysis or equipment malfunction failures) previously performed. The change of the setpoint for the unloading valve for the waste gas compressor does not change the method of compressing the gas. These changes do not alter the interface of the waste gas compressors with systems important to safety. The changes affect only a non-safety grade system which has no accident mitigation function.

The changes do not create any new accidents or equipment malfunction failures, and they do not reduce the margin of safety as identified in the Technical Specifications.

Therefore, on the basis of the evaluation of effects, it can be concluded that the proposed change is acceptable from a nuclear safety standpoint and no unreviewed safety question exists.

SA-SE Number *WBPLEE-97-080*

Affected Documents	*Document Type*	*Safety Assessment Title*	*Implementation Date:*
Westinghouse NSAL 96-004 N3E-934 Exception Number 53 EX-N3E-934-53	Procedure	Safety System/Control System Interaction	8/20/96

Description of Change, Test, or Experiments

Westinghouse advised TVA through Nuclear Safety Advisory Letter (NSAL) 96-004 of a potential problem involving a shared instrument tap on each of the steam generators (SG).

NSAL 96-004 SYNOPSIS
The original reactor protection system design included both a SG low-low water level reactor trip function and a low feedwater flow reactor trip function. The low feedwater flow reactor trip was included in the design to provide a backup reactor trip to the low-low level trip. This backup trip was required to address an IEEE 279-1971 standard requirement relative to control system and protection system interactions which could result from potential SG level channel failures. The SG low-low level reactor trip function is implemented with three narrow range instrument channels combined in a 2 out of 3 logic configuration. The output from one of the level channels is also used to control SG level. When a protection channel is also used for control purposes such that a single failure of the protection channel results in an undesirable control system action which may require a subsequent protection system action, Section 4.7.3 of IEEE 279 requires that an additional random protection system failure be assumed when applying failure criteria. Accordingly, if the level channel performing the control function were to fail, this standard requires the assumption of a second level channel failure. With two failed level channels, the 2/3 logic for low-low level reactor trip could not be satisfied. The low feedwater flow reactor trip feature was installed as a backup protective trip for low steam generator level transients to satisfy the control system/protection system interaction requirement of IEEE 279. Addition of a fourth level channel would also satisfy the requirement.
The low feedwater flow reactor trip was eliminated by DCN M-16617-C as part of the Eagle 21 process protection system upgrade. In conjunction, in order to satisfy the control system/protection system interaction requirement of IEEE 279, a median signal selector module was installed for each SG to isolate the SG level control function from the protection function. Each signal selector module compares the three SG level signals and selects the median value for control system use. By rejection of the high and low signals, the control system is prevented from acting on any single failed protection channel. Since failure of a single protection channel will not result in an adverse control action with this arrangement, the assumption of a second level channel failure is not required, the 2/3 logic will satisfy single failure criteria, and the low feedwater flow trip is not required.
The failure of the common SG level/steam flow instrument tap was not considered in the elimination of the low feedwater flow reactor trip. Failure of this tap would result in a low steam flow signal coupled with a high SG level signal in one channel. The control function of the steam flow channel would begin to close the feedwater control valve on low steam flow and reduce steam generator level. Since the reduction in SG level represents an adverse control system action which is not mitigated by the median signal selector module, IEEE 279 requires assuming the failure of a second redundant SG level channel. This second failure would prevent operation of the low-low level reactor trip protection feature since the 2/3 trip logic could not be satisfied.

TVA documented the condition in WBPER960760. Corrective action for the PER consists of technical justification to show this is not a credible event at WBN. This SA/SE is the documentation of the technical justification established by WBN Engineering.

Safety Evaluation Summary

This SA/SE does not constitute a change to the facility but an engineering evaluation that justifies no additional changes. The WBN technical justification has three levels of justification which are incredible circumstances, detectable event, and correctable event. Each of these levels will be discussed. The three are presented as a "defense in depth" in that any one evaluation justifies the position of "use as is". However, the defense in depth demonstrates a broader focus in problem recognition and correction.
TVA WBN determined the failure of the common sense line or common process tap to be incredible for the following reasons. First, the sense line and tap are passive elements designed to meet all seismic and thermal stress conditions without exceeding material limitations. This is reiterated in WB-DC-40-31.50 Pipe Break Criteria in that postulated pipe breaks are not assumed in pipe one inch or smaller. Second, the sense lines are all welded % inch schedule 80 SS pipe from the tap on the SG to the instrument panel. Third, equipment vibration and thermal movement of the SGs are isolated from the sense line by means of a qualified flex hose welded in line. And lastly, Westinghouse characterized this as a rapid transient. In a simulated failure of the steam flow transmitter and the level transmitter, over 2 minutes elapsed between the first indications and plant trip due to low-low SG level. During this time the annunciator system pointed the operator to the problem for rapid diagnosis and mitigation. Discussion of these points follows.
The first point is the sense line and tap are passive devices. The sense line is supported to seismic cat I criteria and analyzed so that under all conditions pressure boundary integrity is maintained. This criteria was implemented under the TVA construction program and ASME Section III installation program.
Second, postulated pressure boundary breaks can be limited to areas of high stress. Thus, areas of low stress can be excluded as potential break areas. The sense line is a 1/2 inch schedule 80 pipe of all welded construction. The Westinghouse standard quoted in the NSAL utilized 3/8 inch tubing with compression fittings joining the tubing lengths. The compression fittings present a stress riser where the ferrule "bites" the tubing. And, compression fittings have a history of failure due to vibration. Thus WBN differs from the standard reviewed by Westinghouse.
Third, the normal forces that induce high stresses are not present in this configuration. Normal high stresses are caused by thermal movement and vibration causing flexure of the piping. Both of the forces are isolated by means of a flex hose qualified for 2500 psi service at 650 F temperature through millions of vibration cycles. The flex hose is an all metal construction with inconel bellows and stainless steel ends and braiding. The flex hose is not subject to the degrading effects of radiation since all metallics are used. The installation instructions and inspection criteria assure the flex hose will work within the design parameters of the SG. Vendor testing of the hose showed a burst pressure of over 11,000 psi. This greatly exceeds the potential pressures of the SG and therefore provides adequate margin that the flex hose will not be a failure point. The pipe itself is designed to hold much higher pressures than the SG conditions i.e. 1200 psi at 600 F SG conditions versus 1826 psi at 650 F pipe rating. At actual 100% power conditions of 542 F and 980 psi, the allowable working pressure of the pipe would be about 1900 psi.
The second level of defense is the detectability of the event. The event was modeled on the simulator by forcing the steam flow channel low and the level channel high. Annunciators and main control room indicators lead the operator to the problem. From the time the event started and

left unattended to plant trip, was 2 minutes and 4 seconds. This is adequate time for the operator to recognize the problem and move to the correction or mitigation phase.
The third level of defense follows the second level in that the event is correctable. The end point of the event will be a successful plant shutdown. The first action the operator would probably take is placing the level control system in manual. This eliminates the interaction of control and safety system, and stabilizes the SG level. With the threat of an immediate trip reduced, the operator can concentrate on determining root cause, i.e. broken sense line. This should be apparent by raising temperatures and humidity in containment also alarmed in the control room. With a cause established, the operator has the option of manually tripping the plant or going through an orderly controlled shutdown. This of course would depend on factors at the time of the event.
Finally, in letter WAT-D-8057, Westinghouse stated the common sense line used by the redundant channels in Reactor Vessel Level Indicating System (RVLIS) did not compromise redundancy since it was inconceivable the common tap would break because it was in a protected area. This is one phase of the justification given in this SA/SE.
Therefore, the evaluation explains why the present design does not increase the probability or consequences of an accident identified in the SAR, nor increases the probability or consequences of malfunction to equipment important to safety, nor creates an accident or malfunction different from those evaluated in the SAR, and lastly does not reduce the margin of safety as defined in the basis for tech specs. These statements can be made by applying the three defense in depth areas to the problem. Thus the event becomes incredible, is detectable and is mitigable.

SA-SE Number *WBPLEE-97-089-0*

Affected Documents	*Document Type*	*Safety Assessment Title*	*Implementation Date:*
DCN M-39293-A FSAR Change Pkg. 1473 TRM Change Pkg. 97-002 -TRM Table 3.3.1-1	DCN	Cycle 2 Design Basis Analyses Parameters	9/12/97

Description of Change, Test, or Experiments

This Safety Assessment/Evaluation addresses changes in Cycle 2 design basis analyses parameters and changes in instrumentation response and control characteristics and setpoints as discussed in Subsections a & b. These changes provide more operational margin as well as operational enhancements in reactor control and protection.. The design basis analyses which support these changes also satisfy commitments made to the NRC to provide consistency between analysis parameters used for LOCA and non-LOCA analyses in the reactor vessel upflow conversion (WCAP-11696 and WCAP-11627) and to address 10CFR50.46 modeling issues. The respective TVA letters to NRC are dated July 28,1993 "Watts Bar Nuclear Plant (\NBN) - Emergency Core Cooling System (ECCS) -Evaluation Model Changes (TAC Nos M86069 and M86070)", LBLOCA NCO 920041002 & -03 and letter dated August 28,1995 "Watts Bar Nuclear Plant (WBN) - Emergency Core Cooling System (ECCS) -Evaluation Model Changes (TAC Nos M86069 and M86070)", SBLOCA NCO 920041006.

In addition, the Cycle 2 design basis analyses parameters account for a longer fuel cycle (greater than 12 months but less than 18 months). Note that the changes associated with the 18 month fuel cycle are covered in a separate Safety Assessment/Evaluation. Since the safety analysis input parameter changes associated with the 18 month fuel cycle are included in the analyses which support this safety evaluation, the SA/SE of the changes associated with the 18 month fuel cycle can be performed separately without affecting the conclusions of this SA/SE.

The safety analysis assumptions and input parameter changes and the various plant hardware setpoint changes evaluated in this SA/SE are implemented via DCN M-39293-A, FSAR Change Package 1473, and TRM Change package 97-002 and include:

a. Setpoint/Scaling Changes-DCN M-39293-A

· For Cycle 2 and later revise the OT-T/OP-T reactor trip setpoints by adjusting gains and lead/lag time constants to enhance operating margin and incorporate tolerances for variations in indicated loop delta T and Tavg in the AT/Tavg and Low SG level Trip Time Delay (TTD) channels.

· Revise setpoint and scaling uncertainty analyses to support Cycle 2 and later operation with 10% SGTP and additional 2% RTDF and provide the basis for a TS change to allow operation at reduced flow, if it becomes necessary to take advantage of the additional margin.

· Revise the control system power mismatch non-linear gain breakpoint from 1 % to 2% to further minimize rod stepping due to power oscillations which can be caused by lower plenum anomaly.

· Revise Pressurizer Pressure SSDs to revise compliance value for DUB limit.

· Revise SG Level Control Point from 66.5% down to 60% to reduce moisture carryover.

Some of the changes require modification to the plant Technical Specifications and have been

Safety Evaluation Summary

· The setpoint/scaling changes do not increase the probability of an accident.

The proposed changes do not result in a condition where the design, material and construction standards, which were applicable prior to the changes, are altered. The OTdeltaT and OPdeltaT setpoints/scaling changes do not require any physical hardware changes and are used in the various defined analyses for accident definition/mitigation.

All of the affected NSSS systems and components have been evaluated with the Total Design Flow (TDF) of 93,100 gpm. The primary loop components (reactor vessel, reactor internals, CRDMs, loop piping and supports, reactor coolant pump, steam generator, and pressurizer) meet the applicable structural limits with the revised TDF of 93,100 gpm and will continue to perform their design functions. The RCCA drop time remains unaffected and the current design core bypass flow remains valid.

All of the applicable acceptance criteria for the accidents described in the FSAR continue to be met.

· There is no increase in the consequences of an accident: The SLB radiological doses are unaffected and are still within the existing licensing basis limits.

· The proposed changes do not cause an increase in the probability of an accident. The changes in the tolerances for deltaTO, T' and T" also do not require any physical hardware modifications and only require changes to the Technical Specification Allowable Values for the OPdeltaT and OTdeltaT setpoints and for the vessel deltaT equivalent to power functions. Thus, there is no increase in the probability of an accident since the appropriate Allowable Values have been modified to determine channel operability for these functions.

· The proposed changes do not cause the initiation of any accident nor create any new limiting single failures. The OTdeltaT and OPdeltaT protection functions are used for accident mitigation and do not initiate any accidents. Also, the affected systems and components will still perform their intended design functions.

· The proposed changes do not create any new failure modes for safety related equipment. The changes do not result in any original design specification, such as seismic requirements, electrical separation requirements or equipment qualification being altered.

· The margin of safety for the applicable safety analyses has not been reduced. All of the applicable DNB limits continue to be met for the non-LOCA analyses. The LBLOCA input parameters do not require adjustment for the TDF of 93,100 gpm. The SBLOCA has been re-analyzed for the TDF of 93,100 gpm, and the SBLOCA PCT is well below the 2200°F limit. The affected NSSS systems and components will still meet the applicable design limits and perform their intended safety functions with the TDF of 93,100 gpm. Also, the SLB and LOCA M8`E addressed in TS Change Package 96-013 as submitted to the NRC (see discussion relative to these in the following text and Reference 2q).

In addition the issues of Westinghouse Technical Bulletin ESBU-TB-96-07 and the WBN response are addressed. This bulletin addresses the impact of hot and cold leg streaming and changes in hot leg streaming due to burnup-dependent radial power redistribution on indicated delta T (delta To), delta T span, and the uncertainty calculations for the OT delta T and OP delta T reactor trip functions. The WBN response to this bulletin committed to implement bulletin recommendations, including verification of the analysis limits for hot and cold leg temperatures and evaluating the effects of operation at reduced full power loop delta T values on the OT delta T and OP delta T uncertainty allowances.

b. Analyses Changes-FSAR Change Package1473 and TRM Change Package 97-002

1. Steam Generator Tube Plugging (SGTP) and Reduction in Thermal Design Flow (RTDF)
2. Rod Control System Optimization
3. Reduced Feedwater Temperature at 0 to 25% Power
4. Increased Spray Initiation Delay Time for Containment Integrity
5. Increased OT-T/OP-T Response Time from 7 to 8 Seconds
6. Reduced Steam Generator Water Level from 66.5% to 60%
7. OT delta T and OP delta T Reactor Trip Setpoints Margin Enhancement

c. LOCA and SGTR Analyses

1. Large Loss of Coolant Accident (LBLOCA)
2. Small Break Loss of Coolant Accident (SBLOCA)
3. Steam Generator Tube Rupture (SGTR)

d. Non-LOCA Related Analyses: The Non-LOCA analyses have been performed for the following plant changes.

Cycle 2 design features (e.g. - FQ = 2.5, FÁH = 1.60)*
Steam generator tube plugging (SGTP) of 10 %
Reduced thermal design flow (RTDF) of 2 %
OT delta T/OP delta T margin enhancement
OT delta T/OP delta T response time increase
Rod Control System Optimization (revised setpoints)
Reduced feedwater temperature at 0 to 25 % power
Reduced steam generator water level at 100 % power

releases are still within the applicable equipment qualification limits. The SGTR doses remain within the applicable 10 CFR 100 limits, and the steam generator margin to overfill is maintained.

SA-SE Number *WBPLEE-97-103-0*

Affected Documents

TRM Change Pkg. No. 97-006
TR 3.3.5
TRM Revision 6

Document Type

Technical Requirements Manual and Bases

Safety Assessment Title

Throttle or Governor Valve Inoperable in Open Position

Implementation Date:

9/8/97

Description of Change, Test, or Experiments

Drawing Deviation DD 96-0070 was initiated to identify that the depiction of piping through the steam chests for the HP turbine was not correctly shown on 1-47W610-1-3A, 1-47W610-47-2 and 1-47W611-1-2. DCN S-39212-A revised 1-47W610-1-3A, 1-47W610-47-2 and 1 -47W611-1-2 to correctly depict the steam chest piping. Safety Assessment (SA) WBPLEE-96-209-0, which was completed for DCN S-39212-A, did not identify that TRM Section 3.3.5 and FSAR Section 10.2.4 required change. WBPER970262 documented the inadequate SA and the corrective action plan for WBPER970262 revises these documents. The TRM is being revised on TRM Change Package No. 97-006 and this Safety Assessment/Safety Evaluation (SA/SE) WBPLEE-97-103-0. The affected FSAR section is being addressed by FSAR Change Package 1486 and WBPLMN-97-067-0.

This change revises the Technical Requirements Manual (TRM) (Change Package No. 97-006) to revise actions A1., A.2 & A.3 for Technical Requirement (TR) 3.3.5 (Turbine Overspeed Protection). TR 3.3.5 requirement is provided to ensure that the turbine overspeed protection instrumentation and the turbine speed control valves are operable and will protect the turbine from excessive overspeed. The first alternative action A.2 allowed unlimited time for operation following closing the valve which was considered in series with the inoperative valve. The 78 hours allowed another 6 hours above the 72 hours in condition A.1 (repair valve). Specifically, action A.2 was based on assuming that steam flow through the chest was a series flow of steam entering a Throttle (T) valve and going to only one governor (G) valve (as shown in Figure A). However, the steam flow through the steam chest is such that steam can enter through a T valve and can exit through either G valve (as shown in Figure B).

The current Action A.2 statement includes "close at least one valve in the affected steam line". Using Figure A, an inoperable (failed open) T or G valve would require the T or G in the same "steam line" to be closed. This action insured that overspeed protection was afforded for the HP Turbine with an inoperable valve. However, due to the crossover flow which occurs in each steam chest (Figure B), an inoperative (failed open) T or G would require the closure of both T's or G's to afford overspeed protection. The turbine is designed to have steam from at least three G valves entering the turbine blades for as much full arc as possible. Therefore, operation with steam from only one chest would not be an option.

The HP turbine and associated throttle and governor valves are not safety related and are not required to perform a primary or secondary safety function. The closure time of 0.15 seconds remain unchanged for these valves. All turbogenerator protective trips that will automatically trip the turbine (mechanical) and generator (electrical) abnormalities remain unchanged. There are no plant modifications created by this TRM change.

Based on the above, Technical Requirements Manual (TRM) (Change Package No. 97-006) revises actions A.1, A.2 & A.3 for Technical Requirement (TR) 3.3.5 (Turbine Overspeed Protection).

Action A.1.1 requires Operations to determine the operability for the two high pressure turbine steam inlet valves on the steam chest which are opposite the inoperable valve. Action A.1.2 requires Operations to restore the inoperative valve to operable status. Operability is established by either having a fully functional valve or manually closing the otherwise inoperative valve. Action A.2.1

Safety Evaluation Summary

The TRM change revises the alternative actions for TR 3.3.5 that were based on an invalid assumption that a series steam flow path through a specific throttle and governor valve set could be blocked to restore overspeed protection and thus would have allowed unlimited run time with the inoperative flow path blocked. This change relies on restoring overspeed protection.

This TRM change does not reduce Operations ability to respond to an inoperative high pressure turbine steam inlet valve. The requirement that turbine inlet valves be capable of isolating the main turbine from the steam supply even if one of the valves fails open is unchanged. The changes in the TRM reflect a different means to achieve the same goal. The goal of the second action is unchanged in that if there is a valve stuck in the open position the steam flow path through that valve will be terminated by closing other valves. The changes reflect that, due to the chest configuration, closing the steam path through the affected valve requires the unit be taken off line.

Since there is no change to the plant, there is: (a) no increase in either the probability of occurrence of an accident previously evaluated in the SAR or probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR, (b) no increase in either the consequences of an accident previously evaluated in the SAR or the consequences of a malfunction of equipment important to safety previously evaluated in the SAR, (c) no possibility for the creation of either an accident of a different type than any evaluated previously in the SAR or malfunction of a different type than evaluated previously in the SAR and (d) no reduction in the margin of safety as defined in the basis for any Technical Specifications.

TRM Change Package 97-006 is acceptable from a nuclear safety standpoint and no unreviewed safety question exists.

requires Operations to determine the operability for the two high pressure turbine steam inlet valves on the steam chest which are opposite the inoperable valve. Action A.2.2 requires total steam isolation to the high pressure turbine in the event that the inoperative valve cannot be closed. Action A.3 provides another option to Operations to isolate the turbine from the steam supply by using the Main Steam Isolation Valves (MSIV). This alternative assumes there may be partial leakage through the high pressure steam inlet valves.

SA-SE Number *WBPLEE-97-154-0*

Affected Documents	*Document Type*	*Safety Assessment Title*	*Implementation Date:*
DCN W-39742-A	DCN	Fire Protection Report	12/12/97

Description of Change, Test, or Experiments

The subject DCN corrects an Appendix R control interaction that could have caused the loss of both control room air handling units A-A and B-B (respectively 0-MTR-31-12-A and 0-MTR-31-11-B) and circulation pumps (1-MTR-31-80/1-A and 1-MTR-31-96/1-B) providing chilled water to the AHUs. The control circuits for both of these air handling units are routed in the same fire zone such that a fire could cause the control power fuses to blow for both air handling units resulting in their loss. The subject DCN is adding manual switches to the control circuits for each air handling unit that can be used to transfer the essential control circuits to a different set of fuses as a manual action in case of a fire. This new set of fuses, switch, and the associated wiring are contained totally within the 480V C & A Bldg Board such that control is selected for the new configuration, and potential Appendix R interactions are eliminated. The addition of these control stations ensures that no possible Appendix R failure mechanism results in the air handling units not be available.

Safety Evaluation Summary

This change does not affect any FSAR evaluations (accident analysis or equipment malfunction failures) previously performed. No new accidents or equipment malfunction failures are created. The Technical Specification is not affected. This change is in compliance with safety classification requirements as specified in design basis documents. The revised manual action identified in the Fire Protection Report replaces an existing manual action, with the action occurring at another location. The time that the action has to occur remains unchanged. Therefore, on the basis of the evaluation of effects, it can be concluded that the proposed change is acceptable from a nuclear safety standpoint and no unreviewed safety question exists.

SA-SE Number *WBPLEE-97-168-0*

Affected Documents	*Document Type*	*Safety Assessment Title*	*Implementation Date:*
FSAR Change Pkg. 1498 Hydro WO 98-06103-00	FSAR	Watts Bar Hydro 161 kV Switchyard to be Removed From Service	1/28/98

Description of Change, Test, or Experiments

The proposed activity is the deletion of Bus Splitting Relays from the Watts Bar Hydro Plants 161 kV switchyard and the required change to Watts Bar SAR as identified in Watts Bar SAR Change Package No. 1498. These relays were originally installed to provide assurance that the Watts Bar Hydro Plant Generators would remain stable under postulated worst case fault conditions.

Corporate Nuclear Engineering and Transmission/Power Supply (TPS) have identified and agreed that the worst case fault conditions (a three-phase fault and simultaneous stuck breaker) that had been postulated in transmission system studies were both unrealistic and beyond what was required. These positions are well documented in memorandum from TVA's Chief Electrical Engineer (B43 920310 001) dated March 6,1992, which specifies the credible grid contingencies that should be utilized to study offsite power requirements and a supplement to this memo dated June 25,1997 (E32 970625 600).

TPS no longer considers a three-phase fault and simultaneous stuck breaker as part of their planning criteria. TPS now considers a phase-to-ground fault with a stuck breaker for the worst case scenario.

Transmission Planning Department's most recent study documents that the bus splitting relay scheme is not required to maintain the Hydro Plant generators stable under postulated fault conditions of a phase-to-ground fault with a stuck breaker. This study shows that offsite power supply voltage recovery is significantly improved if the automatic bus split does not occur. Therefore, the Transmission Planning Department is proposing that the automatic bus split relay scheme used at the Watts Bar Hydro Plant 161 kV switchyard be disabled by permanently lifting of wires on these relays and opening trip cutout switches or PK blocks on the associated trip circuits.

Although there is not a clear mechanism for a new failure mode and one is certainly not expected, the worst case failure mode scenario of the proposed activity that could be hypothesized would be the complete loss of offsite power. This scenario is adequately enveloped by FSAR section 15.2 9 which addresses the accident analysis for a complete loss of all offsite power coincident with the loss of onsite AC power to the station auxiliaries. This is addressed as a condition 11 (Faults of Moderate Frequency) event. The deletion of the bus splitting relays at the Watts Bar Hydro plant for its 161 kV switchyard will have no impact on the accident analysis that has been performed to address the much broader issue of the complete loss of offsite power source coincident with the loss of onsite AC power. The loss of the offsite power supply is also adequately controlled by T/S section 3.8.1 for operating modes 1-4 and T/S section 3.8.2 for shutdown modes 5-6 during movement of irradiated fuel assemblies.

Safety Evaluation Summary

This proposed activity does not present an unreviewed safety question as the disabling of the bus splitting relays at the Watts Bar Hydro plant does not diminish the capability or capacity of the 161 kV offsite power requirements as imposed by GDC 17. Based on issued transmission system studies, the deletion of these relays will actually be an improvement to the offsite power system in that voltage recovery is significantly improved if the automatic bus split does not occur. The accident analysis in the SAR addresses the complete loss of offsite power coincident with the loss of onsite AC power. The current SAR analysis is bounding for the worst possible results that could be postulated from this proposed activity and this activity will not result in any new accidents or malfunctions of a type than what has been previously analyzed. A review of the T/S sections 3.8.1 and 3.8.2 which address loss of offsite power as well as the past NRC SERs has not identified any margin of safety or acceptance limits which would be affected by this proposed activity.

SA-SE Number *WBPLEE-98-001*

Affected Documents	*Document Type*	*Safety Assessment Title*	*Implementation Date:*
FSAR Change Pkg. 1502	FSAR	Removal of References Having No Relevance	1/23/98

Description of Change, Test, or Experiments

In reviewing Chapter 8 of the FSAR, several references to figures were identified as having no relevance to the general subject. FSAR Change Package 1502 removes those references.
- Section 8.2.1.2, "Transmission Lines, Switchyard, and Transformers"
The paragraph that starts with "The location of the common station service transformers A" discusses the location and physical separation of transformers. The section references Figures 8.2-3 and 8.2-6A. Only one, if any, reference is necessary. Figure 8.2-6A is an example of related conduit details with no obvious connection to text specifics.
- Section 8.2.1.3, "Arrangement of the Start Boards, Unit Boards, Common boards, and Reactor Coolant Pump (RCP) Boards"
Three paragraphs of Section 8.2.1 -3 are involved. They begin with 1) "CSSTs C and D are connected to 6.9-kV common switchgear C and D", 2) "From the 6.9-kV start board the two 6.9-kV start buses....", and 3) " The four unit station service transformers are located...." Briefly, these paragraphs discuss the 6.9-kV common-to-shutdown board cable routing method and note that cables are routed in conduit banks and cable trays. Reference is made to Figure 8.2-6A and 8.2-6B. The specific feature of importance is not identified, and the figures provides no clarification or amplification of text.
- Section 8.2.1.8, Conformance with Standards
The two paragraphs of Section 8.2.1.8 that start with 1) "The 6.9 kV common switchgear C and D are connected to the 6.9-kV shutdown boards by cable arranged to provide two physical independent sources of offsite power." and 2) "A chain fence (see Figure 8.2-6A)...." discuss the distances of conduits from buildings and landmarks. The referenced figures show detail cuts of conduit and cable trays. These figures do not identify the subject features, i.e. the fence, and do not provide understanding of the text.

Figure numbers and titles:
8.2-6 Electrical Equipment Common and Unit Station Service and RCP 6900V Buses Sheet 1, TVA Drawing No. 45N230- I RE
8.2-6A 500 kV Transformer Yard Conduit & Grounding Common Switchgear C & D Conduit Bank Plan & Details, TVA Drawing No. 75W802-7 R6
8.2-6B Turbine Building Unit 2 Conduit & Grounding Cable Trays - Elevation 728.0 Plan & Details, TVA Drawing No. 45W880-4A RD
8.2-6C Auxiliary Building - Unit I Conduit & Grounding Elevation 757.0 Details, TVA Drawing No. 45W828-18 R15
8.2-6D Auxiliary Building Conduit & Grounding Elevation 757.0 Details, TVA Drawing No. 45W828-17 RH

Also noted, Docketed Correspondence 1984 Docket, January 06, 1984, 8401110.130, Watts Bar Nuclear Plant (WBN) Safety Analysis Report (FSAR) Amendment references FSAR Figures 8.2-6A, -6B, -6C, and -6D. SER including supplements does not discuss or reference these figures or their subject, i.e., cable numbers, plan view, detail view, etc.

Safety Evaluation Summary

FSAR CHANGE PACKAGE 1502 contains no change that results in a modification to the facility, procedures, testing, Technical Specifications, maintenance activities, commitment to NRC, and does not constitute an unreviewed safety question.

Regulatory Requirement of Section 8.2
Per RG 1.70 requirements, FSAR Section 8.2.1 is to describe the offsite power system with sufficient detail to demonstrate (1) functionality of structures, systems, and components important to safety. Circuits that supply power for safety loads from the transmission network should be identified and should to meet GDC 17 and 18. Voltage level and length of each transmission line from the site to the first major substation that connects the line-to-grid should be provided as well as unusual features.

10CFR50, Appendix A, Criterion 17 requires an adequate electrical offsite, onsite, and batteries power system be discussed including capacity, capability, specified acceptable fuel design limits, independence, and failure modes. In short, an adequate, diverse power system is to be demonstrated.

10CFR50, Appendix A, Criterion 18 requires periodic inspection and testing requirements for operability and functional performance of important areas, i.e., wiring, insulation, connections, switchboards, power source, relays, switches, and buses under the various operating conditions.

SA-SE Number *WBPLMN-95-086-0*

Affected Documents	*Document Type*	*Safety Assessment Title*	*Implementation Date:*
DCN S-37649-A	DCN	Flood Protection Requirements Issued as Design Critria	11/27/95

Description of Change, Test, or Experiments

The reference DCN issues design criteria WB-DC-40-29,R6, "Flood Protection Provisions," as a design output document. The design criteria is issued as a design output to provide flood protection requirements in a format that can be used by the plant organizations to prepare/revise AOI's and other plant procedures. This revision of the design criteria incorporates comments from operations and maintenance organizations.

Safety Evaluation Summary

Flood warnings are issued in time to permit the reactor to be shutdown in an orderly manner. Other design bases events are not postulated concurrent with a PMF. Flood protection features were developed considering appropriate redundancy to accommodate equipment failure modes. Therefore, the proposed activity does not create any new or previous conditions or malfunctions of equipment important to safety that were not previously analyzed in the FSAR. The Technical Specifications do not define a margin of safety for flood mode operations. Margins of safety for equipment associated with other design bases events are not applicable to flood mode protection. Therefore, on the basis of the evaluation of effects, it can be concluded that the proposed change is acceptable from a nuclear safety standpoint and no unreviewed safety question exist.

SA-SE Number *WBPLMN-95-110-1*

Affected Documents	*Document Type*	*Safety Assessment Title*	*Implementation Date:*
TACF No. 1-95-013-3	Temporary Alteration	Removal of Internals from Check Valve 1-CKV-003-0821 at the Discharge of the 1B-B Motor Driven Auxiliary Feedwater Pump	10/3/95

Description of Change, Test, or Experiments

The change proposed by TACF 1-95-013-3 is to remove the internals from check valve 1-CKV-003-0821 at the discharge of the 1B-B motor driven auxiliary feedwater pump. Testing at low flows has revealed that vibration occurs which may be caused by an interaction of the pump and the discharge check valve. The check valve was noted to be slamming during testing, creating large piping displacements. The removal of the internals from this check valve will allow for testing to determine if the suspected interaction is the root cause of the vibration. If testing is successful, the intent is to make the modification permanent. This safety evaluation is only for the temporary modification and a separate safety evaluation will be required for the permanent modification. This modification impacts the system as described in the SAR since the check valve in question is shown in figure l0.4-21 and Table 10.4-6. Table 10.4-21 assigns the function of maintaining the discharge line inventory. This function will be accomplished without this valve. This function prevents water hammer by maintaining water solid conditions in the discharge line and preventing backflow into the pump, the results of which is a potential to steam bind the pump or overpressurize the suction piping. The discharge line will be maintained in a water solid condition without the subject check valve due to the fact that the water in the condensate storage tank is at a higher elevation then the discharge line up to the level control valve (the portion of the line normally open to the pump discharge). Since the discharge line will be maintained in a water solid condition, water hammer will not be a concern. The function to prevent back flow into the pump of high temperature water and the potential to cause steam binding is prevented by valves 1-CKV-003-921, -861, and -830 for steam generator #3 and valve 1-CKV-003-833 for steam generator #4. Each of the lines to the steam generators from the AFW system has a temperature transmitter which is monitored for high temperature which indicate that back flow is occurring (due to leaking valves). Action is required when high temperatures are noted. In addition each pump has a local temperature indicator at its discharge which is required to be checked on a regular basis. These measures will ensure no steam binding of the pump will occur. This temporary modification does not change or impact any of this equipment or procedures.

Should air be trapped in the discharge line upstream of the level control valve, the potential exists that during testing on recirculation that this air would be compressed and on pump shutdown the expanding air would drive water back through the pump and potentially over pressurize the pump suction. This is prevented from happening by the fact that this section of the piping is below the level in the condensate storage tank and therefore any leakage will not introduce air into the system. The only way air could remain in the system is for the system to be improperly vented. Any opening of the level control valve will clear the air in this line. Even if the air were to remain in the line several vent paths exist in the system. The recirculation line is open to the condensate storage tank and the pump casing is open to the atmosphere through the pump packing. Either path should be able to relieve the small quantity of water required to prevent overpressurization in the unlikely event that some air is trapped and pressurized in the discharge line upstream of 'the level control valves.

Safety Evaluation Summary

Based on the above discussion the function of the check valve will be maintained without the internals being in place. Therefore the function as described in the SAR is maintained even though the description is affected. The answers in section C will show that no unreviewed safety question exists and prior approval of the NRC is not required to implement this temporary change. The basis for this conclusion is that the removal of the check valve internals will not affect the ability of the AFW system to perform its safety function, does not change the function or the method by which the function of the AFW system performs, and does not change the requirements the system has to meet. No change to the Technical Specifications is required for this change.

SA-SE Number *WBPLMN-95-139-0*

Affected Documents	***Document Type***	***Safety Assessment Title***	***Implementation Date:***
DCN S-37766-A	DCN	Baseline Effort for the WBN Emergency Operating Instructions (EOIs)	12/5/95

Description of Change, Test, or Experiments

This SE is a baseline effort for the EOIs and also addresses DCN S-37766-A, which issues calculation WBN-OSG4-188 Revisions 5 through 9 as design output for the EOIs. The EOIs were produced, following the guidelines of the ERGs, as a means for the operator to mitigate plant events by following planned steps, approved by Westinghouse and evaluated by TVA, that ensure safe shutdown of WBN. Calculation WBN-OSG4-188 provides the setpoint values for the EOIs. The development of the EOIs, and issuance of WBN-OSG4-188, does not affect the SAR since the instructions were based on mitigation requirements and processes already addressed in the SAR.

Safety Evaluation Summary

The WOG/ERG procedures were produced specifically to address accident mitigation for Westinghouse-designed plants. The WOG/ERG Rev 1B procedures are the latest approved comprehensive set of symptom-based technical guidance instructions for response to emergency transients. The WBN EOIs were developed directly from the ERGs. Calculation WBN-OSG4-188 provides the setpoint values for the EOIs. The ERGs provide the methodology and strategy for mitigation of a broad spectrum of plant emergencies, both within and beyond the design basis of the plant. The WBN EOIs fully incorporate these strategies and methods for accident response. The ERGs are technically supported by analysis using various computer codes outlined in the ERG's background documentation. The required postaccident safety functions and parameters are ensured to be met. Therefore, the EOIs were developed and approved as a WBN-specific, safe method of operator response to plant events.

SA-SE Number *WBPLMN-95-170-0*

Affected Documents	*Document Type*	*Safety Assessment Title*	*Implementation Date:*
DCN S-37995-A FSAR Change Pkg. 1403	DCN	HEPA Filters in Service Building "Hot Shop"	12/21/95

Description of Change, Test, or Experiments

DCN S-37995-A voids WB-DC-40-24.2 and WB-DC-40-47 as well as DIM-WB-DC-40-47-1 because all rooms and areas addressed by these criteria have been converted to offices, storage and meeting rooms. This change was done on DCN M-16544-B. The HVAC systems (including the HEPA filters were left in place and used to provide the cooling and ventilation needs for the new rooms. The protective clothing will be sent offsite for cleaning. In addition, the "Hot Machine Shop" areas will not be used. If any welding, machining, etc. is needed a portable HEPA filter will be set up to use the equipment in the existing machine shop on elevation 713.0 of the Service Building. Future plans are for a "Hot Machine Shop" and a "Hot Tool Room" in the Auxiliary Building. A "Small Tool Decontamination
Room" is already located in the Auxiliary Building. Since there will be no potential for radioactively contaminated air in these areas, there is no need for HEPA filters in the HVAC systems serving them. The Configuration Control Flow Diagram 1-47W866-6 is therefore revised to remove the HVAC systems for these areas from configuration control in accordance with Watts Bar Business Practice BP-354, Appendix C, paragraphs B and C. The "as-designed" flow diagram 47W866-6 is revised to include these HVAC systems and to note that the systems may be operated with or without the filters.

The only change to the SAR is to delete the Protective Clothing Decontamination Facility (and the ventilation room which was part of the area served by the PCDF HVAC system), and the machine shop and small tool and equipment decontamination area as potential sources of radioactively contaminated air, in the evaluation of the monitoring requirements for the Service Building Vent. (See FSAR Section 11.3.8).

Safety Evaluation Summary

Since there is no increase in potential for exposure to radioactively contaminated air, there will be no increase in the offsite dose or ALARA concerns for personnel in these areas. A USQ does not exist for the changes made by DCN S-37995-A because the probability or consequences of an accident or equipment malfunction is not increased by the change. In addition, the change does not create the possibility of an accident or malfunction of a different type than any previously evaluated in the SAR and the margin of safety defined in the Tech Specs is not reduced.

SA-SE Number *WBPLMN-95-172-0*

Affected Documents	*Document Type*	*Safety Assessment Title*	*Implementation Date:*
FSAR Change Pkg. 1405	FSAR	WBPER950636 identified a condition where design basis documents did not agree with the FSAR.	1/24/96

Description of Change, Test, or Experiments

WBPER950636 Rev.0 and Rev. 1 identified an adverse condition where design basis documents did not agree with the FSAR. These problems were corrected on DCN S-38582-A and FSAR Change Package 1404. One additional condition was identified as part of the extent of condition evaluation. Section 6.4.3 of the FSAR states that the doors in the Main Control Room Habitability Zone (MCRHZ) are security doors with card-operated electric locks. It also states that the control room operators will be able to respond to an alarm in the MCR if one of these doors is opened. However, the security system has been modified and some of the doors that form the MCRHZ pressure boundary no longer have card-operated locks. The alarms are no longer located in MCR. However DCN M-22223-A modified the MCRHZ pressurization system and added pressure switches that monitor the pressure in the MCRHZ with respect to the surrounding areas and trip alarms if the positive pressure is lost. These alarms are used by the MCR operators to determine which doors are open or the location of the breach to the pressure boundary. None of the operating procedures rely on the security system alarms for this function. FSAR Change Package 1405 corrects the method used to monitor the MCRHZ pressure boundary. No equipment is modified and the function of all components and the system remains the same. Therefore, this is a documentation change only.

Safety Evaluation Summary

The changes to the methods used to monitor the pressure boundary integrity of the main control room habitability zone will have no adverse impact on the SAR accident analysis. This change is not a physical change. No equipment has been modified or had its function changed. The control room operators will be better able to monitor the MCRHZ pressure boundary integrity during normal operation using the pressure switches than using the security door alarms. Since the control room pressure will be better monitored for leaktightness, the system response to any accident will be more reliable, therefore, the consequences of an accident will not be increased.

This change is only a documentation change and does not change the function of any system or component. Therefore, there will be no possibility of creating a new type of malfunction with this change. The Control Building pressure requirement in the Technical Specifications section 3.7.10 will not be changed. Therefore, the margin of safety will not be affected by this change.

A USQ does not exist for the changes made in the FSAR because the probability or consequences of an accident or equipment malfunction is not increased by the change. In addition, the change does not create the possibility of an accident or malfunction of a different type than any previously evaluated in the SAR and the margin of safety defined in the Tech Specs is not reduced.

SA-SE Number WBPLMN-95-175-1

Affected Documents	Document Type	Safety Assessment Title	Implementation Date:
TRM TABLE 3.7.5-1-Rev. 2 TRM-95-095 FSAR Change Pkg. 1408	Technical Requirements Manual and Bases	Minimum Temperatures in the Main Steam Valve Vault Rooms	1/4/96

Description of Change, Test, or Experiments

The Technical Requirements Manual, Table 3.7.5-1, Items 18 and 19, North and South main steam valve vaults (MSVV) are revised to change the lower acceptance monitoring normal and abnormal limits from 80 degrees Farhrenheit (F) to 50 degrees F and the area defined is clarified to state "at affected MSSVs." Since temperature monitoring in the MSVVs is limited to ensuring operability of the MSSVs, this clarification is warranted to ensure that only the MSSVs are addressed by this instruction. TRM Bases B 3.7.5, Background is also revised to add discussion on the effect of ambient temperature on MSSV setpoints as discussed. Due to valve design, ambient temperatures can affect the setpoints of the main steam safety valves (MSSV), whereby a decrease in valve temperature causes an increase in setpoint, resulting in a non-conservative relief pressure. Ambient temperatures within the MSVV are monitored each shift by Technical Requirement Instruction (TRI) 1-TRI-30-1, Balance of Plant Temperature Monitoring Program, which ensures that the MSSVs minimum temperature is maintained to meet the plus/minus 1 percent code allowable on setpoint pressure. Only the lower limit is monitored since an increase in pressure, which is conservative.

The change revises the normal minimum temperature in the MSVV from 80 degrees F to 50 degrees F. This change agrees with the current MSSV temperature limits of 50 degrees F to 150 degreees F as curently defined in the Main Steam System Description. This decreased lower temperature limit permits a lower acceptance temperature limit for the MSSVs for operability determinations.

Safety Evaluation Summary

It is concluded that the change in the SAR Section 0.4.3.3.7, which generally describes those systems which maintain the steam valve vault temperature, does not affect any safety-related function and is bounded by the existing FSAR Chapter 15 safety analyses which assumes the MSSV setpoint tolerance to be plus/.minus 1%. Additionally, the MSSVs as defined in Technical specification 3.7.1 are only required to be operational during Modes 1, 2, and 3. Therefore, the proposed change to the FSAR does not represent an Unreviewed Safety Question pursuant to 10 CFR 50.59 (a)(2).

SA-SE Number *WBPLMN-96-005-0*

Affected Documents	*Document Type*	*Safety Assessment Title*	*Implementation Date:*
DCN S-38661 FSAR Change Pkg 1411	FSAR	Provide Suction Flow Path from RWST to Suction of the Safety Injection Pump	3/1/96

Description of Change, Test, or Experiments

The system description for the safety injection (SI) system is revised to delete the requirement to remove power from flow control valve, FCV-63-5 during normal operations. The safety function of FCV-63-5 is to be open to provide a suction flowpath from the RWST to the suction of the SI pumps. The change to allow restoration of power is supported by the Failure Modes and Effects Analysis for the SI system which is in FSAR Table 6.3-8. This analysis demonstrates that spurious closure of FCV-63-5 is not credible. The main control room handswitch is provided with a protective cover to prevent operator error. In addition, the handswitch and transfer switch are wired with contacts on both sides of the motor contactor to prevent a single failure within the switchgear from spuriously closing the valve. These features eliminate the need to remove power from FCV-63-5. FSAR section 7.6.6 requires revision to delete FCV-63-5 from the list of valves which are described as having power removed.

Safety Evaluation Summary

A USQ does not exist for the change to delete the requirement to remove power from FCV-63-5 because the probability or consequences of an accident or equipment malfunction is not increased by the change. In addition, the change does not create the possibility of an accident or malfunction of a different type than any previously evaluated in the FSAR and the margin of safety defined in the Technical Specifications is not reduced.

TVA letter to NRC dated October 27, 1981 identifies valves in the engineered safety features systems (ESFAS) which require power lockout. Valve 1-FCV-63-5 is not included in that list of valves for which TVA committed to lockout power. The initial NRC safety evaluation report (SER) dated June 1982 Sections 7.6.4 and 7.6.6 do not identify valves specifically by unique identifier number, however the basis for the NRC conclusion of acceptability was the FSAR that existed at the time and the October 27, 1981 letter. None of the SER supplements have revisited the subject of the Sections 7.6.4 and 7.6.6. The original WBN design maintained power on this valve until 1993. That change was considered an enhancement to the system. Therefore, the change to restore power to 1-FCV-63-5 does not have the potential to affect the conclusions stated in the SER, and do not modify a previous commitment to the NRC.

SA-SE Number *WBPLMN-96-015-2*

Affected Documents	*Document Type*	*Safety Assessment Title*	*Implementation Date:*
DCN W-38663-A FSAR Change Pkg. 1422	FSAR	Gate Valve Subject to Thermal Binding and Bonnet Pressurization	10/22/96

Description of Change, Test, or Experiments

Revision 5 of the calculation, "Evaluation of Gate Valves Subject to Thermal Binding and Bonnet Pressurization" demonstrates that 1-FCV-1-16 is capable of opening, if required, following a design basis accident with the maximum pressure trapped in the bonnet. The design basis accident which has the potential to trap the highest pressure in the bonnet is a main feedwater line break (MFLB). If the MFLB occurs on steam generator #1, 1-FCV-1-16 is required to open to provide steam from steam generator #4 to the auxiliary feedwater pump turbine (AFWPT). The limiting MFLB is a full double-ended rupture outside containment between the check valve and containment. Based on the WBN Design Criteria, for Major Rupture of a Main Feedwater Pipe, all feedwater is assumed to be discharged out the rupture and the levels in all of the steam generators begin to decrease. Eventually the main isolation valves (MSIV) close, terminating reverse steam generators subsequently increases until the main steam safety valves open at 1185 psig, plus 3% setpoint tolerance and 3% accumulation yielding 1256 psig. This is the pressure that becomes trapped in the bonnet of 1-FCV-1-16. After the main steam safety valves lift, steam generator pressure steadily falls (FSAR Figure 15.4-13C). The resulting system/valve parameters when the valve is required to open are 1256 psig trapped in the bonnet, 485 psig in the intact upstream piping, and 100 psig in the downstream piping. These are the most severe operating conditions under which the valve could be required to open. The above calculation shows that a margin of 1.6% exists between the available valve operator thrust and the thrust required to open the valve. However to improve the margin for valve opening, a hole was drilled in the upstream disc.

Safety Evaluation Summary

An unreviewed safety quest does not exist for the design change to drill a hole in the upstream disc of 1-FCV-1-16. The disc hole will equalize pressure between the bonnet and the upstream piping and will eliminate the possibility for trapping pressure within the bonnet. Accordingly, the probability or consequences of an accident or equipment malfunction is not increased by the change. In addition, the change does not create the possibility of an accident or malfunction of a different type than any previously evaluated in the FSAR and the margin of safety defined in the Technical Specifications is not reduced. This design change actually increases margin for valve opening if the AFW system is required for accident mitigation. Drilling of the upstream disc hole on 1-FCV-1-16 also decreases the probability of a malfunction or accidents previously evaluated in which pressure locking prevents the valve from opening. This is accomplished by increasing the margin between the operator's capability and the thrust required to open the valve. Therefore, this design change only serves to enhance the operability of the valve.

SA-SE Number *WBPLMN-96-016-1*

Affected Documents

DCN S-38724-A

FSAR Change Pkg. 1421

Document Type

FSAR

Safety Assessment Title

Personnel Access Doors and Hatches Which are ABSCE Boundaries Designated With a Sign

Implementation Date:

3/4/97

Description of Change, Test, or Experiments

DCN S-38724-A corrects the following documentation to agree with the plant configuration: system description N3-30AB-4001 to add requirements that ABSCE doors and hatches be appropriately labeled; design criteria WB-DC-40-59 clarifies that (1) door/door sets DE1/DE4 and DE1/DE5 are air locks which are not electrically interlocked, and (2) that the ERCW pipe tunnels hatches are part of the ABSCE boundary; and the FSAR was revised to clarify which ABSCE air lock doors are electrically interlocked.

Safety Evaluation Summary

The Auxiliary Building Secondary Containment Enclosure (ABSCE) doors must be closed to ensure the Auxiliary Building Gas Treatment (ABGTS) and the Emergency Gas Treatment System (EGTS) maintains a slight negative pressure in the ABSCE compared to the outside environment. This negative pressure will minimize out leakage of potentially radioactive gases to the outside environment in accordance with 10 CFR 20 or 100 limit guidelines. A review of the detail changes leads to the SE conclusions that this documentation revision is safe and does not constitute an USQ.

The design criteria, system description, and FSAR revisions defined above have no impact on the FSAR accident analyses since these are building doors that have infrequent egress/ingress through them during an SI/ABI. These doors, and all ABSCE doors, shall be posted to designate these doors as ABSCE Boundary Doors, Keep Closed. The only doors that should be used during an SI/ABI for entry into the Auxiliary Building are those from the Radiochemical Lab and Service Building both on elevation 713. These documentation changes have no impact on a malfunction of equipment important to safety. The applicable FSAR Chapter 15 events have been reviewed and this change does not increase the probability of occurrence or the consequences of an accident. This change does not create the possibility for an accident of a different type than any evaluated previously in the FSAR. The ABSCE doors do not involve any credible equipment failure modes, and are not postulated to fail.

The safety function is that the ABSCE doors must be closed to ensure the ABGTS and the EGTS maintains a slight negative pressure in the ABSEC compared to the outside environment. This negative pressure will minimize out leakage of potentially radioactive gases to the outside environment in accordance with 10 CFR 20 or 100 limit guidelines. Ibis change does not create the possibility for a malfunction pathway of a different type than any evaluated previously in the FSAR. Sections 3.3.8 (AEGIS), 3.6.9 (EGTS), 5.7.2.3 "Offsite Dose Calculation Manual", and 5.7.2.7 "Radioactive Effluent Controls Programs" are not impacted by this change. The margin of safety as defined in the Tech Specs has not been reduced.

SA-SE Number *WBPLMN-96-022-4*

Affected Documents

DCN M-38762-A
FSAR Change Pkg. 1427

Fire Protection Report Change Pkg. FP05

Document Type

FSAR

Safety Assessment Title

Pressure Pulsations in Pump Discharge Piping at Low Flow Rates

Implementation Date:

8/15/96

Description of Change, Test, or Experiments

Routine surveillance testing of the motor driven auxiliary feedwater pumps (MDAFWP) has identified an operating condition in which pressure pulsations are experienced in the pump discharge piping at low flow rates. The pulsations are attributable to the low flow capability (approximately 38 gpm) of the existing minimum flow line when the pumps are operated at dead-head conditions. The existence of the pulsations indicates that the 38 gpm minimum flow rate is not sufficient to prevent undesirable recirculation flows in the pump. DCN M-38762-A installs an intermediate capacity (approximately 170 gpm) recirculation line for each MDAFWP to prevent pump operation at 38 gpm. The recirculation line will allow pump operation further out on the pump curve when the demand for auxiliary feedwater flow to the steam generators is low, thereby
eliminating the discharge pressure pulsations.

Each new pump recirculation line contains a pneumatically operated flow control valve (FCV) which can be operated from the main control room. The valves are designed to be normally closed and fail closed upon loss of electrical or pneumatic power. During normal plant operation, the valves can be opened for performance of functions such as ASME Section XI pump testing. However, the control circuit design for the FCVs automatically closes the valves if they are open when any of the signals which automatically start the MDAFWP is generated. The operator has the option of reopening the FCV after a 5 second time delay if the demand for auxiliary feedwater flow to the steam generators is low. Then after the signal which initiated automatic MDAFWP start is reset, if another signal is subsequently generated, the valves once again close automatically. This assures that the forward flow requirements for the purpose of reactor decay heat removal are satisfied

The FSAR is impacted by: (a) the addition of the recirculation line FCVs to the list of valves required to be active for design basis events, (b) the inclusion of the FCVs in the failure mode & effect analysis demonstrating that they will not prevent the auxiliary feedwater system from performing its accident mitigation function, and (c) the addition of text to the auxiliary feedwater section to provide a description of the recirculation lines and associated FCVs.

Safety Evaluation Summary

DCN M-38762-A adds new recirculation piping, controls and valves to the MDAFWPs. This line is being added to reduce pressure pulsations, reduce maintenance, increase pump seal life, and increase reliability caused by low flow conditions during periodic surveillance of the motor driven pumps. A review of the detail changes leads to the SE conclusions that this modification is safe and does not constitute an USQ. The equipment being added does not prevent existing safety-related equipment from performing their safety function. The controls and equipment have been designed to ensure that the safety function of the AFW system to provide the necessary flow to the SG has not been compromised. This change does not create any unacceptable equipment failure modes. No single failure of a component added by this change will cause the system to fail to fulfill its functional requirements. The designed redundancy of the AFT system precludes a single failure from increasing the probability of an accident discussed in the SAR. The components are designed and installed to function during and after seismic events and are qualified to meet applicable environmental qualification requirements of 10 CFR 50.49, thus, no common mode failures are postulated that could affect the electrical independence of the MDPs subsystems and the TDP subsystem. The AFT is used during accidents to provide SG cooling water and to prevent primary to secondary lube leakage and subsequent release of fission products. This modification does not change the logic, function or design basis of the system with regard to the delivery of water to the SG. Any failures of the added components are bounded by existing FSAR transients. This change does not reduce the margin of safety identified in the basis for any Technica1 Specification.

SA-SE Number *WBPLMN-96-026-0*

Affected Documents	*Document Type*	*Safety Assessment Title*	*Implementation Date:*
DCN M-38825-A FSAR Change Pkg. 1428	FSAR	Temperature Setpoints for Condensate Demineralizer Polishers	5/3/96

Description of Change, Test, or Experiments

The function of the condensate polishing demineralizer system (CPDS)is to remove dissolved and suspended impurities from the secondary system which includes the condensate/feedwater systems. The removal of impurities and corrosion products in the secondary system reduces corrosion damage to the secondary system equipment. The CPDS also removes impurities which might enter the system through the makeup water, and removes radioisotopes which will be carried over to the secondary cycle in the event of a primary-to-secondary steam generator tube leak. Currently the maximum operating temperature, as controlled by setpoints, of the CPDS is 130°F as identified in reference 2.4. The current design temperature for portions of the system is 120°F. However, DCN M-38825-A will revise the design temperature to 140°F (acceptable per calculation EPM-HRP-010894) and revise the maximum Instrument setpoint temperature to 135°F to permit greater operating flexibility. The limitation for this change is the maximum recommended operating temperature of the Dow Chemical resins which is stated in a Dow letter dated April 26, 1996 to be 140°F. The setpoint temperature for various CPDS instruments will be revised by this DCN to permit additional operating flexibility.

It is recognized that the only operating limitation which indirectly affects this DCN will be the main turbine back-pressure limitation of 5.5 in HgA with a saturation temperature of 137°F, above which the potential for turbine blading damage exists. The operator will receive an alarm at 5.0 in HgA and manually trip the turbine at 5.5 in HgA. These operating setpoints and functions remain unchanged.

The FSAR is revised by DCN M-38825-A to reflect an increase in the operational limits of the CPDS. Safety Evaluation Report (June 1982), section 10.4.6, (including supplements 1 through 20) are not impacted because the operational limits are not addressed.

Safety Evaluation Summary

The system is not required to serve any primary or secondary safety functions and therefore is not required for safe shutdown of the reactor. Therefore, a USQ does not exist for the DCN M-38825-A change to operational limits because the probability or consequences of an accident or equipment malfunction is not increased by the change. In addition, the change does not create the possibility of an accident or malfunction of a different type than any previously evaluated in the SAR and the margin of safety defined in the Tech Specs is not reduced.

SA-SE Number *WBPLMN-96-027-0*

Affected Documents	*Document Type*	*Safety Assessment Title*	*Implementation Date:*
FSAR Change Pkg. 1430	FSAR	Power Ascension Test Program (PATP) 100% Specified Power Levels Defined in Various Ways	1/17/96

Description of Change, Test, or Experiments

TVA has generally followed the Westinghouse sequence document, WAT/WBT-SU-3.1.9 "Test Sequence at 100% Power" which provided the guidance for increasing power to 100% and the sequence for performing full power tests. It is not unusual for the site specific procedure (PATs/PETs) to allow a "window" to the extent practicable for performing the tests. Typically, the full power plateau is described as 100% +0/-2 which permits operating from 98 to 100% when performing the applicable tests. If a test is to be run at other than the specified power level (e.g., 100%), the results from tests that are a function of power level must then be extrapolated from the measured power under which the specific test is conducted to the value at 100%.

The initial startup testing program for WBN Pressurized Water Reactor (PWR) is governed by Regulatory Guide (R.G.) 1.68, R2 dated August 1978. Appendix A of the R.G., titled "Initial Test Program," paragraph 5, states "Parenthetical numbers following the items listed below indicate the approximate power levels for conducting the tests.", e.g., In subparagraph a. power levels required are shown as (25%, 50%, 75%, 100%), with other single/multiple power level requirement combinations in various other subparagraphs through o.o. WBN Power Ascension Test Program (PATP) for various tests at 100% power have generally specified power levels in accordance with R.G. 1.68 R2. However, in the Power Ascension/Escalation Tests descriptions in FSAR Chapter 14. 2 (Reference 12), 100% has been defined in various ways, e.g., "approximately 100%, "100% power level," "100% power," and "100% plateau." The respective PATs/PETs as listed below, contain prerequisite statements such as "100%" power level or 100% plateau" where 100% power tests are to be conducted, which is then defined later in the test procedure as being between 98-100%. The purpose of the proposed change is to address various operation conditions at WBN as well as meeting current power requirements for the TVA grid, yet meet the requirements of the PATP testing to be completed at the 100% plateau (i.e., 100 % RTP, et. al.). The change defined in the FSAR Change 1430 and respective changes to six specific PATP tests is to redefine/and justify the "100% power level or 100% plateau" for specific PATs/PETs (references 2 through 7) as being for an applicable range of 95-100% versus the present 98-100%. In order to support these changes, justification has been provided by Westinghouse to perform the various full power tests at as low as 95%. PATs referenced previously as being addressed by this Safety Evaluation are:

1-PAT-1.7 R1/CN-14 "Operational Alignment of Process Temperature Instrumentation".
1-PAT-8.6, R1/CN-2 "Plant Trip from 100% Power (Turbine Trip)".
1-PAT-3.3, R1/CN-3 "RCS Flow Measurement".
1-PAT-8.0, R1/CN-4 "Test Sequence for 100% Plateau".
1-PAT-1.3, R2 "Large Load Rejection Test"
1-PAT-1.2, R1/CN-3 "Load Swing Test"

Safety Evaluation Summary

Review of the Westinghouse evaluation of the proposed change and justification for conducting 100% PATP power tests at a power level as low as 95% indicates that the specific PATs/PETs designated to be conducted at 100% power level (or 100% plateau) can be done versus the previously defined range of 98-100% power. However, for the 5 specific PATs/PETs being addressed by this Safety Evaluation, in addition to the Westinghouse summary type description of the tests, they have provided a discussion of any impact associated with performing the test as low as at 95% power and the actions that should be taken before ascending to 100% power. In the current definition of 98-100% power level, as long as the tests are conducted within that domain the caveats of evaluating data taken at the 95% power level to assure control systems settings are adequate for the 100% power level do not need to be applied. Conducting the tests at 95% power level requires that the Westinghouse instructions be followed in order that the 100% power level parameters upon which the safety analysis of record is based will not be violated as well as placing the reactor in an unsafe operating condition.

Some of the evaluations outlined in the Westinghouse letter, can adequately be addressed by TVA (e.g., 1-PAT-1.7 or 1-PAT 3.6 "Incore Moveable Detectors" which is addressed in the Westinghouse letter but is not required to be addressed in this Safety Evaluation). However, for the Large Load Rejection Test (1-PAT-1.3) Westinghouse must be appraised of the data taken at 95% power level in order to determine if the control system settings are adequate, i.e., based on the transient from 95%, project the results as if the test had started from 100% to verify that the acceptance criteria are met. The Westinghouse evaluation is also equally applicable to 1-PAT-1.2 "Load Swing Test" which already defines the power level as 95-100% for 100% power level (100% plateau), however is included as part of this Safety Evaluation because of the Westinghouse conditions associated with running the test at as low as 95% power level. For the Plant Trip From 100% Power (1-PAT-8.6) Westinghouse has confirmed that the acceptance criteria for this test are applicable to the test performed from a power level as low as 95%.

Based on the Westinghouse evaluation and directions for extrapolation of data taken at 95% to assure control system settings are adequate for operation at 100% addressed either by TVA or as required by them the proposed change is considered safe and will not affect the Safety Analysis of Record.

SA-SE Number *WBPLMN-96-029-0*

Affected Documents	*Document Type*	*Safety Assessment Title*	*Implementation Date:*
DCN W-38725-A	DCN	Spent Resin Storage Tank Backflush Water Connection	9/20/96

Description of Change, Test, or Experiments

Sequoyah experienced difficulties in draining the Spent Resin Storage Tank (SRST) due to resin caking in the bottom of the tank. This DCN W-38725-A adds a 2 inch backflush water line to the Spent Resin Storage Tank (SRST) drain line to resolve this issue. The new piping and valves that are being installed should dislodge agglomerated resins in the tank drainage. The backflush water is controlled by the manual diaphragm valve O-ISV-077-0695A with remote valve operator, and a check valve O-CKV-077-0696A is added to prevent SRST drainage flow from entering into the Primary Makeup Water System.

The new line will be supported by a new rigid type (y,z) pipe support. The design for this support is also included in this DCN. Modifications to existing valve and piping supports were necessary as a result of valve gallery access. The Civil calculations demonstrate that the piping and pipe supports affected by this DCN are qualified in accordance with all applicable requirements for all design loading conditions.

The SRST room is considered a "Very High Radiation Area" with dose > 500 Rad at 1 meter. A flood level detector (O-LS-040-0044) that is in the SRST room alarms in the main control room (MCR) to alert the operator that there is a pipe leak in this room. This LS will be moved just outside the rooms at the remote valve operator station, to prevent an individual from receiving an excessive dose during maintenance or calibration activities. The Electrical portion of this DCN provides the routing for the cable and conduit to move the flood level detector to outside the SRST room.

The FSAR impact associated with DCN W-38725-A is that the new resin drain backflush line will be depicted on flow diagram 1-47W830-3 which is FSAR Figure 11.2-1 (sheet 3) and Figure 11.5-1. There are no other FSAR impacts.

Safety Evaluation Summary

DCN W-38725-A adds valves and piping that enable the operators to backflush the drain piping in the SRST. It does not change the system design basis, logic or function of any system that is important to safety. The modification does not increase the probability of occurrence of an accident. Nor does it increase the probability of occurrence of a malfunction of equipment important to safety. A review of the detailed changes leads to the SE conclusions that this modification IS safe and does not constitute an USQ. There are no credible failure modes that would cause the Solid or Liquid Waste Disposal System to be unable to perform its function of processing spent resin. Nor does this change affect any equipment failure modes that are important to safety. The piping and valves being added by this change are located in the Auxiliary Building. The piping from the bottom of the SRST to the new isolation valve may contain radioactive materials and, therefore, will be installed Seismic I and will conform to the ASME code requirements. This system is not used for any accident mitigation function. A SRST rupture is bounded in the FSAR by the existing analysis for a Waste Gas Decay Tank Rupture FSAR Section (15.3.5). This chance does not reduce the margin of safety identified in the basis for any Technical Specification. The addition will not prevent any component from performing its function as described in the Technical Specifications.

SA-SE Number *WBPLMN-96-034-0*

Affected Documents	*Document Type*	*Safety Assessment Title*	*Implementation Date:*
FSAR Change Pkg. 1434	FSAR	Radiation Zone Map Figures Deleted and FSAR Chapter 12 Revised	6/20/96

Description of Change, Test, or Experiments

Radiation zone map figures 12.3.18, 12.3-18a and 12.3-19 in calculation WBNTSR-077 are based on design drawings 46W421-3, and 46W425-7. These drawings are no longer maintained under the drawing configuration control process. Consequently, the figures do not correctly depict the current plant configuration. Figures 12.3.18, and 12.3-19 are included in chapter 12 of the FSAR. Figure 12.3-18a is not included in the FSAR. The FSAR figures are intended to show Radcon facilities such as the Health Physics Lab, personnel decontamination station, Health Physics count room, etc. The Radcon facilities are currently described in FSAR Chapter 12.5.2. This change is to delete the subject figures from calculation WBNTSR-077 and the FSAR, and to add statements to FSAR sections 12.3.1 and 12.3.2 to refer to section 12.5.2 for information on Radiological (Radcon) equipment, instrumentation and facilities. Section 12.5.2 contains all information necessary to meet the intent of RG 1.70.

Safety Evaluation Summary

This change is a documentation change only. Deletion of the subject FSAR zone maps in lieu of reference to an alternate FSAR section does not increase the probability of occurrence, or the consequences of an accident or equipment malfunction previously evaluated in the SAR, and does not create a possibility for an accident or malfunction of equipment of a different type than any evaluated previously in the SAR. In addition, the change does not reduce the margin of safety as defined in the basis for any Technical Specification, since the zone maps are not discussed in the Technical Specifications nor Technical Specification Basis.

SA-SE Number *WBPLMN-96-035-2*

Affected Documents	*Document Type*	*Safety Assessment Title*	*Implementation Date:*
DCN M-38817-B FSAR Change Pkg. 1433	FSAR	Stub-outs and Isolation Valve Installation of Condenser Bypass lines on No. 2 Feedwater Heaters	10/15/96

Description of Change, Test, or Experiments

A condition has been identified at Watts Bar in which the #2 feedwater heaters do not drain to the #3 heater drain tank during rapid load reductions at low power operating conditions. The #2 feedwater heaters are at a lower elevation than the #3 heater drain tank. At lower power levels, #2 extraction steam coming into the #2 feedwater heaters is not at high enough pressure to force condensate up to the higher elevation of the #3 heater drain tank. If level in the heater reaches the Hi-Hi setpoint, condensate through the respective heater will isolate. Intermediate pressure heater string isolations have occurred previously during a rapid load reduction which was required on 3/27/96 when reactor coolant loop NO-3TAVG dropped below 551°F due to a failure of the no. 3 reactor coolant pump to transfer from the alternate to the normal power supply bus. This event is documented in PER NO. WBPER960210 and LER 50-96/96-011.

Condenser bypass lines similar to that which currently exists for the #1 feedwater heaters will be installed to correct the problem. The new #2 feedwater heater bypass lines will bypass condensate to the condenser when level exceeds the normal control level of the existing #2 heater level control valves which drain to the no. 3 heater drain tank. DCN R-38794-A installed the stub-outs and isolation valves necessary for future installation of the #2 feedwater heater condenser bypass lines. This approach allows installation of the bypass lines and control valves under DCN M-38817 without penetrating the pressure boundary of any portion of the heater drains and vents system that is required for normal plant operation. The stub-out connections to the system 6 pressure boundary, the stub-out connections to the #2 feedwater heater level trees, and the stub-out connections to the control air supply are all included within the scope of previous DCN R-38794-A. The scope of Advance Authorized DCN M-38817-Rev. A included installation of the level control valves and associated piping, and connection to the existing stub-outs provided previously by DCN R-38794-A. Rev. B of DCN M-38817 issues the full scope of the change by adding the level indicating controllers (LICs), the control air supply, position indication for the LCVs in the MCR, field fabrication of a steel platform at the LCVs, and thermal qualification of the bypass piping, including field fabrication of spring hanger supports near the LCVs. The full scope of DCN M-38817-Rev. B is addressed by this SA/SE.

This SA/SE also addresses revision of SOI-5 & 6 for the scope of adding any valves associated with DCN M-38817-B to the valve alignment checklists.

Safety Evaluation Summary

The failure modes which must be considered for addition of the No. 2 feedwater heater condenser bypass lines by DCN M-38817-B are (a) an open failure and (b) a closed failure of the new level control valves. (a) The new condenser bypass LCVs will fail closed upon loss of control air. Only the failure of a single bypass LCV at any given time requires evaluation. A common mode failure mechanism which could open all three LCVs simultaneously does not exist because each LCV is independently controlled by the LIC from its respective heater level tree. (b) A simultaneous closed failure of all the condenser bypass LCVs could occur on loss of control air. A closed failure of the new condenser bypass valves would simply cause the system to function as if the bypass line did not exist (i.e., prior to DCN M-38817-B implementation). Accordingly the change is safe and no USQ exists.

SA-SE Number *WBPLMN-96-037-0*

Affected Documents	*Document Type*	*Safety Assessment Title*	*Implementation Date:*
FSAR Change Pkg. 1435	FSAR	Procedure change clarifies 50% step load reduction	6/26/96

Description of Change, Test, or Experiments

Chapter 14 of the FSAR is being revised to more accurately describe the Large Load Reduction Power Ascension Test. This test subjects the plant to an approximately 50% load decrease from the 100% power test plateau. The FSAR describes this decrease as a "step decrease". Actual test conduct resulted in the decrease occurring in two phases as a result of automatic Balance of Plant (BOP) control system actuation. The required licensing basis change modifies Table 14.2-2, sheet 35 to add a Note describing the 2 phase reduction in load observed during the conduct of the test.

The large load reduction test demonstrates the ability of the primary and the secondary side systems, including the automatic control systems, to sustain a 50% decrease in turbine generator load. The automatic control systems tested are the rod control system (non-safety related), steam generator level control system (safety related), steam dump system (non-safety related), and the pressurizer pressure/level control system (safety related). The requirement to perform the test can be found in Regulatory Guide 1.68, revision 2, Appendix A, Section 5, paragraph hh, which states "Demonstrate that the dynamic response of the plant to the design load swings for the facility, including step and ramp changes, is in accordance with the design." TVA implements this requirement in WBN FSAR test abstract Table 14.2-2 sheet 35. TVA's procedure for this test, 1-PAT-1.3, Large Load reduction satisfies these FSAR requirements. Acceptance criteria for these tests are the reactor and turbine do not trip, there is no safety injection, the pressurizer safety valves do not lift, steam generator safety valves do not lift and the plant can achieve stable conditions without manual intervention.

The test began by initiating test prerequisites on May 10, 1996 and was complete on May 12, 1996. The procedure was performed from the 100% test plateau as directed by 1-PAT-8.0.

The load reduction was initiated by entering a calculated setter value based on the turbine operating power value (100.7%). The new setter value (47.3%) was entered into the Electro-Hydraulic Control (EHC) panel and the load rate was set to 200%/min. The "GO" button on the EHC panel was then depressed and the load reduction was initiated. During the reduction the EHC system toggled from IMP IN (actual turbine load feedback) to IMP OUT (fixed governor system position) mode and stopped the transient. The value shown on the load reference was 61.5%. Approximately 2.5 minutes later the No. 3 Heater Drain Tank bypassed to the condenser and initiated a Balance of Plant BOP runback to approximately 48% power. Data collection equipment continued to collect data until the Test Director determined that stable conditions had been achieved.

Using turbine generator load the first step was approximately 27%. The turbine generator load then slowly increased until the second reduction was initiated. The second reduction was approximately 26% for an overall load reduction of 49%. The response of turbine impulse pressure was more significant. TREF which is generated from impulse pressure was reduced from 586.9 (95.8%) to 572.4 (49.4%). This calculates to an initial drop in load of 46.4%. TREF then increased to 579 (70.5%) until the second runback occurred. After the second runback TREF stabilized to 571.9 (47.8%). This calculates to a total load reduction of as determined by TREF to 48% which is equivalent to 579 MWe.

The total reduction in load, as calculated by the procedure was 569 MWe. This was 10 MWe less

Safety Evaluation Summary

Since TVA test procedure 1-PAT-1.3 was developed from a guideline procedure (Reference j.) provided by Westinghouse the NSSS supplier for Watts Bar (WBN), after test completion the data was forwarded to them for review due to the two phase load reduction step occurring during the conduct of the test.

Westinghouse initial review (Reference c.) concentrated on the four control systems that were critical to this test acceptance. A final detailed expansion of this evaluation was provided in a Westinghouse letter report.

The evaluations by Westinghouse have been reviewed and concurred upon by TVA, i.e., the transient did not diverge and the plant reached steady state values without any instabilities. If the load reduction had occurred in one step the safety systems would not have been challenged, since the control systems performed as designed and were controlling their respective parameters. The data showed that sufficient margin to trip setpoints also existed. Westinghouse determined that based on engineering judgement and experience from 50% load reduction tests on similar plants, the test results are acceptable.

The initial Westinghouse/TVA Test Review Group (TRG) discussions (telecon) permitted approval of the 1-PAT-1.3 test results. TRG concerns relative to potential controller "wind-up" effects were also adequately addressed, whereby the TRG accepted the test results and forwarded them to the Plant Manager for Approval. In the performance of this SA/SE, further review of the test conditions, results, and Westinghouse conclusions have been evaluated with the conclusion that the test is acceptable and is considered safe, and does not affect the safety analysis of record.

than the review criteria value due to the procedural method used to determine the load reduction. Load had increased approximately 40 MWe after the load value was recorded in the procedure but prior to the initiation of the load reduction. Deficiency Notice (DN) 3 was written to document the calculations using the load value prior to the load reduction (1177 MWe) and the value when stability was achieved (592 MWe). Using these values the overall load reduction was 585 MWe, which does satisfy the review criteria.

The two step load reduction was caused by impulse pressure undershooting LOAD REFERENCE by 20%, causing the controller to switch from IMP IN to IMP OUT. When it switched, the control system stopped the evolution in progress, and returned the load to approximately 61%. Immediately following the LOAD REFERENCE decrease initiation, GOV VALVE DEMAND, GOV VALVE AUTO and GOV VALVE #2 POSITION signals spiked downwards in an attempt to lower-impulse pressure to match the LOAD REFERENCE signal. When the above signals decreased to a 25-30% positive magnitude, impulse pressure had decreased to 50% due to fast valve closure response.

Systems responded within limits and as designed. No plant hardware or procedure changes were required as a result of the test performance. Clarification to the FSAR (Chapter 14, Table 14.2-2 sheet 35) has been made to reflect the test behavior.

The test was subsequently rerun with a 50% load rejection.

SA-SE Number *WBPLMN-96-038-1*

Affected Documents	*Document Type*	*Safety Assessment Title*	*Implementation Date:*
DCN 38911-A FSAR Change Pkg. 1436	FSAR	Condensate Polishing Demineralizer System Restraint Deleted	10/4/96

Description of Change, Test, or Experiments

The function of the condensate polishing demineralizer system (CPDS) is to remove dissolved and suspended impurities from the secondary system which includes the condensate/feedwater systems. The removal of impurities and corrosion products in the secondary system reduces corrosion damage to the secondary system equipment. The CPDS also removes impurities which might enter the system through the makeup water, and removes radioisotopes which will be carried over to the secondary cycle in the event of a primary-to-secondary steam generator tube leak. The system is not required to serve any primary or secondary safety functions and therefore is not required for safe shutdown of the reactor.

Currently, each condensate demineralizer service vessel (CDSV) has the capability of polishing the condensate up to a maximum flow of 17,000 gpm per reactor unit. The maximum polisher flow rate as stated in Reference 2.4 is 3,400 gpm (therefore, normally 5 polishers @ 3,400 gpm = 17,000 gpm). The sixth polisher is on standby. The manufacturer (The Dow Chemical Company) for the resin beads recommends a velocity range of 45 to 60 gpm/ft2 for demineralization for both the cation exchange resin and anion exchange resin (see DCN S-38911-A). This range will support polisher flow rates up to 3,800 gpm and will allow for full flow polishing when condensate flows are greater than 17,000 gpm. Therefore, this DCN revises the maximum flow of 17,000 gpm to 19,000 gpm in Section 2.2.14 of system description N3-14-4002, R2, "Condensate Polishing Demineralizer System".

Safety Evaluation Summary

The CPDS, its associated components, piping, and valves are located in the Turbine Building. The CPDS is non-safety related, installed in a non-seismic structure, is not used during any accident, and normally non-radioactive. The CPDS does have the potential to be radioactive in the unlikely event of a large primary to secondary leak. However, this change does not alter how potential radioactive fluid is processed to and released from CPDS. This DCN does not change the logic or function of any system that is important to safety. These changes are within the existing design basis limitations of the ODCM and therefore, do not represent a change to radioactive release criteria or result in higher discharge concentrations (non-radioactive). A review of the detailed changes leads to the SE conclusions that this change is safe and does not constitute an USQ.

This change does alter the system design from an operational perspective. The CPDS does not interface with any equipment whose malfunction could result in an accident which has been evaluated in the FSAR. This DCN does not change or affect the design basis for any system that is important to safety. No additional components have been added by this change. The existing components, if a malfunction occurs, would not cause a radioactive release in excess of the limits established by 10 CFR 20 and 10 CFR 100 since a release from the CPDS is permitted only when the activity is below the limit as defined in ODCM. No new potential single failures of existing components will occur as a result of this DCN. Neither will this change cause this system or any system important to safety to fail to fulfill its functional requirements. The CPDS, its associated components, and piping do not perform any accident mitigation function. This change does not reduce the margin of safety identified in the applicable Technical Specifications. These changes do not prevent any component from performing its function as described in the Technical Specifications. The ODCM limits for releases from the CPDS are not revised or challenged by these changes.

SA-SE Number *WBPLMN-96-040-0*

Affected Documents	*Document Type*	*Safety Assessment Title*	*Implementation Date:*
TS Bases Change Pkg. 96-008 Rev. 5 Bases 3.6.2 and SR 3.6.2.2 Bases Revision 5	Tech Spec Bases	Containment Air Lock Door Interlock Mechanism Defeat Alarm	7/18/96

Description of Change, Test, or Experiments

The Background discussion for Tech Spec Bases Section B3.6.2 - Containment Air Locks contains a paragraph that states "Each personnel air lock is provided with limit switches on both doors that provide control room indication of door position. Additionally, control room indication is provided to alert the operator whenever an air lock door interlock mechanism is defeated." Drawing 1-45W600-90-3 depicts the circuit which provides control room indication when either the inner door, or inner equalizing valve, or outer door, or outer equalizing valve is open for either the upper or lower containment personnel air locks. The circuit satisfies the Tech Spec B3.6.2 Bases statement that "Each personnel air lock is provided with limit switches on both doors that provide control room indication of door position." However, the existing airlock indication does not provide indication of when both the inner door and the outer door are open, as would be necessary to provide indication of when the interlock mechanism is defeated. Each air lock's door and equalizing valve limit switches are wired in parallel and then connected to a single annunciator window at XA-55-23A on panel 1-M-21. The inconsistency between the TS B3.6.2 statement and the existing annunciation is documented in PER no. WBPER960299. TS Bases Change Package 96-008 deletes the statement "Additionally, control room indication is provided to alert the operator whenever an air lock door interlock mechanism is defeated."

FSAR section 6.2.4.2.3 - Penetration Design discusses the personnel air lock design, door status indication in the main control room, and the mechanical interlock which prevents both doors from being opened simultaneously. However, the FSAR does not contain any statements regarding the existence of indication to alert the operator whenever an air lock door interlock mechanism is defeated. Therefore, TS Bases Change Package 96-008 does not affect any information presented in the FSAR or deviate from the description given in the FSAR.

Safety Evaluation Summary

The change to TS Bases B3.6.2 to delete the statement regarding control room indication of when a containment air lock interlock mechanism is defeated does not affect any FSAR evaluations (accident analysis or equipment malfunctions) previously performed. This indication capability is not essential for assuring containment integrity as demonstrated by this SA/SE. No new accidents or equipment malfunctions are created. The TS Bases change does not impact the associated TS Actions or Surveillance Requirements. This change does not adversely affect any equipment important to safety, either directly or indirectly. Therefore, on the basis of the evaluation effects, it is concluded that the proposed change is acceptable from a nuclear safety standpoint and no unreviewed safety question exists.

SA-SE Number *WBPLMN-96-047-1*

Affected Documents

DCN S-38954-A & W-38956-A
FSAR Change Pkg. 1439

Document Type

FSAR

Safety Assessment Title

DCN S-38954-A revises DC & SD to clarify Condensate Polishing Demineralizer System may be released without dilution flow. DCN W-38956-A adds open control which is not interlocked with the radiation monitor controls.

Implementation Date:

3/26/97

Description of Change, Test, or Experiments

The release pathway for the Condensate Polishing Demineralizer System (CPDS) is normally to the Cooling Tower Blowdown (CTBD) piping and as an alternate to the Turbine Building Station Sump. The CPDS may be released provided the CTBD dilution flow > 20,000 gpm. If flow is < 20,000 gpm passing through the CTBD line, a valve (0-FCV-14-451) in the discharge line from the CPDS is automatically closed. The flow is measured with a flow element (0-FE-27-98) in the lines to the diffuser valves (0-FCV-27-100 and 0-FCV-14-101). A signal that is < 3500 cfs is flowing through the hydro units of the upstream dam shall result in cessation of the CTBD discharge to the river by closing the diffuser valves. The same signal shall open a valve (0-FCV-27-97) to the yard holding pond. The diffusers valves are closed with 0-FE-27-98 measuring zero flow because the hydro units, during the summer, do not operate periodically at night. Therefore, the CPDS can not be released.

DCN S-38954-A permits release of CPDS when the hydro units are not operating provided all required prerequisites are satisfied. This change revises Design Criteria (DC), WB-DC-40-37 "Heat Rejection System," section 3.9.3 and System Description (SD), N3-14-4002 "Condensate Polishing Demineralizer System (CPDS)," section 4 to state that the CPDS may be released with the Cooling Tower Blowdown (CTBD) dilution flow < 20,000 gpm provided the tank is sampled and the activity is < to the Lower Limit of Detection (LLD) as defined in ODCM. (Table 2.2-1). An additional clarification was made to the SD to state that Operations shall (1) have the tank sampled prior to each release when CTBD dilution flow < 20,000 gpm and (2) not make any additions to the tank that is being released.

DCN W-38956-A revises the control logic for valve 0-FCV-14-451. An "OPEN" position is added which is independent of the CTBD dilution flow. Previously, the control switch had only a "CLOSE" and an "AUTO" position. The "AUTO" position was interlocked with CTBD dilution flow and radiation level set points (0-RE-90-225) such that if CTBD dilution flow was < 20,000 gpm or if radiation was above the set point as established using ODCM principles, the valve would close. The "OPEN" position will be interlocked with the radiation level set point only such that CPDS releases can be made with low dilution flow provided overall activity levels for 0-RE-90-225 are below those established in the ODCM.

Safety Evaluation Summary

The CPDS, its associated components, piping, and valves are located in the Turbine Building. The CPDS is normally non radioactive, non safety related, installed in a non seismic structure, and is not used during any accident. The CPDS does have the potential to be radioactive in the unlikely event of a large primary to secondary leak. However, these changes permit a release from the CPDS only when the CTBD dilution flow is < 20,000 gpm and the activity is < to the LLD as defined in ODCM. The activity is verified with sampling and analysis prior to release. These changes are within the existing design basis limitations of the ODCM and therefore, do not represent a change to radioactive release criteria or result in higher discharge concentrations (non radioactive). A review of the detailed changes leads to the SE conclusions that this change is safe and does not constitute an USQ.

Though the operational control logic for 0-FCV-14-451 has been modified, this change does alter the system design from a functional perspective. The prerequisites that have been added assure the liquid to be discharged is non radioactive and that no simultaneous additions are being made to the discharge flow path. This DCN does not affect the design basis for any system that is important to safety. No additional components have been added by this change. A two position hand switch has been replaced with a three position switch. No new potential single failure of existing components has been anticipated to occur. Neither will this change cause this system or any system important to safety to fail to fulfill its functional requirements. This change does not affect the radioactive releases in excess of those established by 10 CFR 20 and 10 CFR 100. This change does not reduce the margin of safety identified in the Technical Specifications 5.7.2.3 or 5.7.2.7.

There is a potential impact on the SER. This potential impact does not affect NRC final conclusions regarding radioactive releases, does not however affect nuclear safety and does not involve an USQ.

SA-SE Number *WBPLMN-96-050-0*

Affected Documents	*Document Type*	*Safety Assessment Title*	*Implementation Date:*
FP06 - Fire Protection Report Change Pkg.	Fire Protection Program	Fire Damper in the Volume Control Tank Room Door	9/25/96

Description of Change, Test, or Experiments

The change package consists of the following:

a. Moving the figures V1-1 through V1-14 and tables 6.1 and 6.2 from part V1 to part 11 of the FPR and renumbering them figures 11-27 through 11-40 and tables 14.8.1 and 14.8.2.

b. Changing the various tables and text throughout the FPR to reflect the changes in a above.

c. Minor changes to figures 11-1 and 11-4.

d. Addition of components to section 14.10 that were not covered by Technical Specifications or Technical Requirements (TS/TR).

e. Added evaluation of fire damper in the Volume Control Tank Room door.

Safety Evaluation Summary

Moving the figures and tables from one part of the FPR to another does not change the information contained on the figures or in the tables. This was done to facilitate use of the report by Operations personnel. The changes made to the figures were minor and did not change the system's function or description. The addition of components to section 14.10 was the corrective action to WBPER960160 and ensures that components required for Appendix R and not included in the TS/TR have appropriately documented testing/inspection requirements and operating requirements. The evaluation of the fire damper in the VCT room door determined that the fire damper was acceptable and this adds the evaluation to the FPR section VII. Therefore, this change package to the FPR is acceptable from a nuclear safety standpoint and no unreviewed safety question exists.

SA-SE Number *WBPLMN-96-051-0*

Affected Documents	*Document Type*	*Safety Assessment Title*	*Implementation Date:*
DCN W-38783-A	DCN	Reroute of the Spent Resin Storage Tank Low Point Drain	9/12/96

Description of Change, Test, or Experiments

There are two issues being addressed by this change. The first is a low point drain from the Spent Resin Storage Tank (SRST) located in the Hot Tool Storage Room. The drain was intended to discharge in a floor-drain which has been covered by a plate. The second issue is debris and silt have the potential for accumulating in the Auxiliary Building floor drains (FDs) that drain to the Auxiliary Building floor and equipment drain sump (ABF&EDS) and in the Reactor Building that drain to either the Reactor Building Floor and Equipment Drain Sump (RBF&EDS) or the Reactor Building Floor and Equipment Drain Pocket Sump (RBF&EDPS). The Reactor Building FDs need filters only during outages when high work activities increase the potential to get debris in the drains. The FDs are a part of the Equipment and Floor Drainage System (E&FDS). The FDs that drain to the ABF&EDS have caused sump level switches to have maintenance requests initiated against them for improper indication or controlling functions. The repairs have indicated silt or debris had coated the level switch mechanisms rendering portions of the switch inoperable.

Design Change Notice (DCN) W-38795-A reroutes the SRST low point drain to an Equipment Drain (ED) that is located within a short distance from the FD that is covered. The ED is a closed drain that is processed to the Tritiated Drain Collector Tank (TDCT). The DCN also adds-a note to the Configuration Control Diagrams (CCDs) 1-47W851-1 and 1-47W852-1 to permit sock filters to be added to FDs as necessary to prevent debris and silt from entering the system. The sock filters shall be monitored and replaced as necessary to prevent radioactive hot spots, crud traps, damage and blockage. The filters in the Reactor Building shall only be used during outages and be removed prior to returning the plant back to service.

Safety Evaluation Summary

The Waste Disposal Systems (WDSs) and E&FDS, their associated components, piping, and valves are located in the Auxiliary, and Reactor Buildings. The E&FDS is also located in the Turbine Building. The WDSs are radioactive, and the E&FDS have the potential to be radioactive. Neither are safety related except for containment isolation valves, which are installed in a seismic structure except for the E&FDS located in the Turbine Building, and the containment isolation valves are only used during an accident. This DCN does not change the logic or function of any system that is important to safety. These systems, associated components, and piping do not perform any accident mitigation function except for containment isolation valves which have not been affected. A review of the detailed changes leads to the SE conclusions that this change is safe and does not constitute an USQ.

There are no accident analysis in the FSAR associated with the SRST low point drain or the E&FDS. The equipment involved in the modification does not interface with any equipment whose malfunction could result in an accident which has been evaluated in the FSAR. The requirements that have been added for the sock filter assure ALARA, drains are not blocked, and they are removed in the Reactor Building before returning the plant to operations. The only equipment that could fail is the pipe itself, and this probability is considered remote. Failure of the SRST low point drain could result in a low level radiation release in the worst case event, which is enveloped and bounded by the accident analysis examined in the FSAR for accidents with a potential release beyond 10 CFR 20 and 10 CFR 100 limits. This DCN does not add any additional or different types of failure modes that have not been addressed in the FSAR Chapter 15. These changes do not reduce the margin of safety identified in the applicable Technical Specifications because the ODCM limits for releases from the waste disposal are not revised or challenged by these changes. These changes do not prevent any component from performing its function as described in the Technical Specifications.

SA-SE Number *WBPLMN-96-057-0*

Affected Documents	*Document Type*	*Safety Assessment Title*	*Implementation Date:*
FSAR Change Pkg. 1444	FSAR	Access to Containment at Significant Power Levels	8/20/96

Description of Change, Test, or Experiments

FSAR change package 1444 is issued to revise FSAR section 12.3.2.2 (Plant Shielding) to delete the statement that personnel access to the lower compartment with the reactor at significant power levels will be prohibited except under cases of extreme emergency, and to reference FSAR section 12.5 (Radiological Control) for access control requirements. In addition, FSAR section 11.4 is revised to clarify that continuous monitoring means continuous monitoring when the monitors are normally operating, in lieu of intermittent sampling or grab sampling. Further, FSAR sections 11.4.2.2.4 and 5.2.7.3.1 are revised to clarify that the upper compartment monitor can be used to monitor the lower compartment if the lower compartment monitor fails. The above changes are made to resolve NRC comments in Inspection 390/96-06.

Safety Evaluation Summary

This change involves revision of the FSAR to clarify operation of the upper and lower containment radiation monitors and correction of FSAR text concerning personnel access to containment. These changes do not affect equipment operational requirements as currently discussed in the FSAR and as required in Technical Specifications 3.4.15 and 5.11 and applicable bases. Consequently, a USQ does not exist for this change because the probability or consequences of an accident or equipment malfunction is not increased by the change. In addition, the change does not create the possibility of an accident or malfunction of a different type than any previously evaluated in the SAR and the margin of safety defined in the Tech Specs bases is not reduced.

SA-SE Number *WBPLMN-96-065-0*

Affected Documents

FSAR Change Pkg. 1445
WBPER960364

Document Type

FSAR

Safety Assessment Title

FSAR change eleminates reference to simultaneously raising the reactor cavity water levels with the lift of reactor pressure vessel head during refueling operations and adds information regarding radiation monitoring and visual inspections for potential RCCA withdrawal during head removal.

Implementation Date:

8/21/96

Description of Change, Test, or Experiments

FSAR Change Request 1445 addresses the corrective action required for WBPER960364 by revising FSAR Section 9.1.4.2.1 to agree with reactor pressure vessel disassembly methodology and refueling cavity water level control contained in Maintenance Instruction MI-68.001 and Refueling Operations procedure GO-7. The change to FSAR 9.1.4.2.1 eliminates reference to simultaneously raising the reactor cavity water levels during the lift of the reactor pressure vessel head, and adds information regarding radiation monitoring and visual inspections to identify a potential RCCA withdrawal during head removal.

Since raising the water level simultaneously with raising the vessel head would preclude the ability to visually inspect for potential RCCA withdrawal with head removal, the existing procedures were written to raise the vessel head approximately 144 inches while monitoring for radiation and to perform visual inspections for potential RCCA withdrawal. As currently written, FSAR 9.1.4.2.1 states that after the vessel head has been detensioned, reactor coolant water is raised in the vessel which overflows to the refueling cavity, the reactor head and the refueling cavity water level are then raised simultaneously. After the water reaches a safe shielding depth, the vessel head is taken to its storage pedestal.

The safe shielding depth is defined in FSAR 9.1.4.3.4 as 10 feet above the active fuel region which correlates to 8 feet 10.875 inches above the fuel assembly (Approximate heights are shown in Figure 1). Since the top of fuel in the vessel is elevation 706 feet 9.3 inches, the safe shielding depth is attained when the vessel water level is above elevation 714 feet 8.175 inches. Since the vessel flange is at elevation 725 feet, the safe shielding depth is reached before the vessel is full and overflows to the refueling cavity area. Therefore, raising the water level simultaneously with the vessel head is not required to maintain a safe shielding depth.

The accident analysis presented in FSAR Chapter 15 was reviewed and no accidents analyzed were identified which would be affected by this change. The closest accident is the fuel handling accident described in Section 15.4.5 and 15.5.6. However this accident assumes rupture of all rods in one assembly and occurs during fuel movement. This evolution occurs after the head is removed and fuel movement begins, and is therefore not impacted.

Safety Evaluation Summary

The proposed FSAR change to Section 9.1.4.2.1 removes the requirement that the reactor cavity water level is increased as the reactor vessel head is lifted. This change is being made to allow the vessel water level to be maintained just below the flange at the pre-cavity flood level, to enable visual inspections for potential RCCA withdrawal as is currently provided in site procedures. The accident analysis presented in FSAR Chapter 15 is not affected since there is no increase in probability or consequences of accidents or equipment malfunctions. The margin of safety is not affected since the proposed change occurs for a plant mode (refueling mode thin which the pressure and temperature limits defined by the Technical Specification do not apply. Based on a review of equipment involved, the possibility of an accident or equipment malfunction was not created since there was no equipment functional or configuration change.

Westinghouse had performed a generic evaluation of a reactor vessel head drop in WCAP-9198, and, although not bounding for Watts Bar, determined that the integrity of the fuel cladding and vessel nozzles and core cooling capability would be maintained. Subsequently in Generic Letter 85-11 the NRC determined that completion of Phase I provided sufficient protection from heavy load drops and that their review of Phase II submittals did not indicate the need for further action. GL 85-11 further determined that a single failure proof crane would be necessary to satisfy NUREG 0612 and that cost analysis indicated such modifications would not be cost effective. SSER 13 confirmed Watts Bars compliance to NUREG 0612 Control of Heavy Loads.

Based on the reviews of FSAR Chapter 15 accident analysis and the determinations stated above and the NRC determinations in GL 85-11, the proposed change does not involve an unreviewed safety question.

SA-SE Number *WBPLMN-96-066-0*

Affected Documents	*Document Type*	*Safety Assessment Title*	*Implementation Date:*
DCN W-38937-A SOI--68.01	DCN	Vent Connection on Primary Water Supply Line to the RCS Pressurizer Relief Tank Added	10/8/96

Description of Change, Test, or Experiments

Operating nuclear plants have used vacuum refill to remove air from the reactor coolant system (RCS) during plant startup. The vacuum refill method significantly reduces the time required to completely fill and vent the RCS by eliminating the multiple reactor coolant pump (RCP) starts and stops needed for dynamic RCS venting. As a result, the RCS vacuum refill can reduce critical path time by approximately one day. In addition by eliminating the multiple pump starts and stops for sweeping and venting, less maintenance is required on the RCP electrical, mechanical, and seal systems.

To support future RCS vacuum refill operations, DCN W-38937-A installs a 3" vent connection on the primary water supply line to the pressurizer relief tank (PRT). The vent consists of a branch off the primary water line with a single 3" manual gate valve and a blind flange connection. The line to which the connection is made is ANSI B31.1 piping and performs no safety function. The added vent connection will not be used when the PRT spray line is in service and; therefore, PRT spray is not affected. DCN W-38937-A is unstaged because the entire modification will be performed in a single stage. Therefore, all future discussion of the modification refers to the vent connector as a whole and no further consideration of staging is necessary.

DCN W-38937-A provides the vent connection to facilitate future RCS vacuum refill, but does not authorize performance of vacuum refill. Actual performance of RCS vacuum refill operations can not occur until the appropriate operating procedures are revised and in place with a supporting nuclear safety assessment/evaluation. Therefore, the revision 0 version of SA/SE WBPLMN-96-066 addresses only the nuclear safety impacts of the existence of the vent connection appendage. This SA/SE will be revised or a new SA/SE will be prepared for the operating procedure revisions to address nuclear safety with respect to performance of RCS vacuum refill.

Safety Evaluation Summary

The DCN W-38937-A change to add a vent connection to the primary water supply line to the PRT does not affect any FSAR evaluations (accident analysis or equipment malfunctions) previously performed. No new accidents or equipment malfunctions are created. Technical Specifications, including the margin of safety as defined in the bases, are not affected. This change does not adversely affect any equipment important to safety, either directly or indirectly. Therefore, on the basis of the evaluation of effects, it is concluded that the proposed change is acceptable from a nuclear safety standpoint and no unreviewed safety question exists.

SA-SE Number *WBPLMN-96-067-0*

Affected Documents	*Document Type*	*Safety Assessment Title*	*Implementation Date:*
DCN S-38978-A FSAR Change Pkg. 1447	FSAR	Heat Load Design for Storage of Spent Fuel Assemblies	1/21/97

Description of Change, Test, or Experiments

WBN Safety Evaluation Report (NUREG-0847) dated June 1982 recognized that if a full core is to be offloaded subsequent to a back to back refueling, that a 60 day decay time limitation exists, i.e.; the SER text states (page 9-5) "The FSAR stated that a pool water temperature of 150°F can be maintained when only one cooling train is in operation following a full core discharge subsequent to a normal refueling. This temperature is based on a 60 day decay period before the fuel assemblies of the full core are placed in the spent fuel pool." In WBN FSAR Amendment 24 the description of the Spent Fuel Pool Cooling System (SFPCS) was revised to indicate that a full core off-load immediately after a normal refueling of the other unit could be supported without pool temperatures exceeding 150°F. Subsequent to this time it was apparent in the industry that spent fuel reprocessing would not be available, and concurrent with TVA increase in spent fuel pit storage capacity, a further change in the FSAR description was made; i.e., In amendment 40 of the WBN FSAR, the description changed to reflect that "with only one cooling train in operation and no additional decay of the core to be loaded into the pool," 150°F could be maintained. In addition the FSAR text was changed to reflect that the system normally handles the heat load from 1/3 of a core freshly discharged from each reactor "plus the decreasing heat load from previously discharged fuel."

In August of 1981, one of the NRC questions regarding the capability of the SFPCS was to provide a response to a question asked at Sequoyah (SQN); i.e., system capability to maintain pool temperatures concurrent with a single failure. The October 1981 response to the NRC indicated that the WBN SFPCS capability to maintain the pool water temperature below 150°F with only one cooling train in operation provided that the design heat load utilized was based on normal 12 month cycles staggered by 6 months, back to back refueling outages, and a full core off-load (with specific MWD/MTU for each region) which had 60 days of decay time from reactor shutdown. The SQN response which was similar to WBN was captured in the their FSAR text, however, was not in section 9.1.3 of the WBN text.

FSAR section 9.1.3.1.1 Spent Fuel Pit Cooling currently states "if it is necessary to remove a complete core from one unit immediately after a normal refueling of the other unit, the system can maintain the spent fuel pool water at or below 150°F with only one cooling train in operation and no additional decay of the core to be loaded into the pool. This statement is incorrect. The design bases for two unit operation will support the following statement: If it is necessary to remove a complete core from one unit immediately after a normal refueling of the other unit, the system can maintain the spent fuel pit water at or below 150°F with only one cooling train in operation and 60 days of decay of the core to be loaded into the pool. Therefore, FSAR Change Package 1447 makes appropriate clarifying changes. The 60 day decay time is captured in the SFPCS system description which is design output.

Safety Evaluation Summary

TVA calculation TI-ECS-48 R3 provides the technical bases for the 60 day delay decay time captured originally in section 2.2.3 and now section 4.0 of the SFPCS system description, and the SER. The FSAR discrepancy appears to have been due to (and limited to) not completely updating the FSAR description in section 9.1.3.1.1 when amendment 45 was issued. The statement was not qualified to indicate that it applied to the original spent fuel pool storage arrangement and that additional limitations were applicable to the increased storage arrangement when the pool contained significant amounts of spent fuel from prior refuelings.

Therefore the change to the FSAR text is a clarification only, and does not change the operational characteristic of the system. The SFPCS system description now captures the correct information in section 4.0 for operator action/limitation to either delay fuel movement until after 60 days or control fuel pin transfer rate to limit the spent fuel pool maximum allowable heat load value. In addition there is consistency with what was already captured in section 2.2.3 text. Also review of the SER and all supplements indicate that the design documents are correct and in agreement, i.e., NRC evaluation as captured in SER acknowledges a 60 day delay decay time prior to off-load of a full core. Review of the Technical Specifications indicates that the proposed FSAR text change does not cause any impact; therefore, there is no affect on the TS. The corrective actions require that appropriate procedures be changed to reflect the SFP heat load limitation; however, the clarification that will be provided does not affect a process or procedure outlined in the SAR. As indicated, previously a review of the SER (NUREG-0847 and Supplements) was conducted, and no conflict exists with NRC understanding of what limitations exist.

This change to the FSAR is a clarification and does not change the operational characteristics of the system. There is no change to any design basis accidents or create any new or anticipated operational transients previously evaluated in the FSAR. There is no physical change to the plant; therefore, no equipment malfunctions are increased nor decreases the margin of safety. Therefore, no unreviewed safety question exists.

SA-SE Number *WBPLMN-96-070-0*

Affected Documents	*Document Type*	*Safety Assessment Title*	*Implementation Date:*
FSAR Change Pkg. 1438	FSAR	Five Unrelated Revisions to the FSAR	8/23/96

Description of Change, Test, or Experiments

This FSAR change consists of several small changes. The first change affects Section 6.2.2.2, page 6.2.2-4 under the paragraph heading of Piping. This change clarifies that the Containment Spray piping contains other flanged joints than the pump connection. Valves and joints that may require disassembly for maintenance are commonly flanged. Piping remains, as it was designed and fabricated, in ASME Code compliance. The second change affects Section 6.4.4, page 6.4-9 and is to delete the statement that there is a 112,500 SCF Argon tank in the yard east of the Control Building. The third change affects Section 10.4.10.2, page 10.4-48 and revises the statement that the main feed pump turbine condenser drains to the No. 7 heater drain tank. It actually drains to the condenser and the FSAR is so revised. The fourth change deletes the filter micron rating on the RCP seal water injection filter. The fifth change clarifies that the power supply to the lighting and communications inside the airlock is not emergency supplied.

Safety Evaluation Summary

The first change described above does not change any ASME Piping Code requirements, nor does it modify any previous commitments. It is a clarification to accurately describe the manner that a piping system is comprised of various parts in order to meet ASME Code requirements. The second change updates the FSAR to post-construction configuration when large volumes of argon gas is no longer needed. The third change is an editorial type correction to the FSAR to align the text with the figures and the associated design basis. The fourth change is required because it has become desirable to replace the seal injection filter with filters having progressively smaller micron rating. The fifth change clarifies that the power supply to the lighting and communications is not emergency supplied. This does not affect any design output and is not a requirement change and does not affect any previous commitments or analysis.

The changes of FSAR Change Package 1438 do not affect any FSAR evaluations (accident analysis or equipment malfunctions) previously performed. No new accidents or equipment malfunctions are created. The FSAR changes do not impact the margin of safety credited in the Tech Specs. These changes do not adversely affect any equipment important to safety, either directly or indirectly. Therefore, the proposed FSAR changes are acceptable from a nuclear safety standpoint and no USQ exists.

SA-SE Number *WBPLMN-96-072-0*

Affected Documents	*Document Type*	*Safety Assessment Title*	*Implementation Date:*
FSAR Change Pkg. 1450	FSAR	Hot M&TE Tool Room Radiation Zone Map	12/11/96

Description of Change, Test, or Experiments

Radiation zone map figure 12.3-12 in calculation WBNTSR-077 currently shows rooms A17 and A18 on elevation 692 as the "Hyperfiltration Demin. Rm" and the "CRT Storage Rm". DCN F-38135-A changed these rooms to the "Hot M&TE Tool Room" and "Hot Tool Room". WBPER960426 documented that the DCN did not change calculation WBNTSR-077 nor the FSAR. Consequently, the FSAR figures do not correctly depict the current plant configuration. This change is to update the subject figures from calculation WBNTSR-077 and the FSAR, and to revise operational procedures SOI 77.01 (to ensure RADCON controls access to room A18 "Hot Tool Room" during TDCT fluid transfer to the mobile demineralizers) and SOI 77.03 (to identify that room A18 has the potential to become a high radiation area during resin flushing).

Safety Evaluation Summary

This change affects the function of two rooms in the Auxiliary Building but does not change the reason the rooms are considered high radiation zones. The procedural changes do not increase the probability of occurrence, or the consequences of an accident or equipment malfunction previously evaluated in the SAR, and do not create the possibility for an accident or malfunction of equipment of a different type than any evaluated previously in the SAR. In addition, the change does not affect the margin of safety as defined in the basis for any Technical Specification.

SA-SE Number *WBPLMN-96-074-0*

Affected Documents	*Document Type*	*Safety Assessment Title*	*Implementation Date:*
DCN S-39015-A FSAR Change Pkg. 1449	DCN	Main Feedwater System Description revised to clarify that operation of the Standby Main Feedwater Pump is not limited to operate only during a nominal 85% Turbine Runback during power operations	10/9/96

Description of Change, Test, or Experiments

The two turbine driven, variable speed main feedwater pumps (TDMFPs) are capable of delivering feedwater to the four steam generators (SGs) under all expected operating conditions. If the unit is operating above 85% guaranteed load and loss of one TDMFP occurs, feedwater flow to the SGs must be restored to 85% guaranteed flow within 20 seconds to prevent a reactor trip. This is accomplished by:

1. Automatic starting of the motor driven standby main feedwater pump,

2. Isolation of the main feed pump turbine condenser associated with the tripped pump, allowing 100% condensate flow to be passed through the active main feed pump turbine condenser allowing maximum power operation of the active feed pump turbine.

3. Acceleration of the active drive turbine to its "high speed stop" speed,

4. Unit load runback is initiated and unit load is decreased to less than 85%.

If-the unit is operating at above 67% but below 85% load, the preceding actions occur except that no unit load runback is required. In the case of a loss of a TDMFP, the SBMFP would already be running, however all other conditions as previously outlined would occur.

In order to facilitate this change in normal plant operation, the impact on the safety analysis must be evaluated prior to making the changes to plant documentation, e.g., Revise note on pegs 2, 9 (b), and 29 of the Main Feedwater System description N3-3A-4002 to clarify that the operation of the Standby Main Feedwater Pump (SBMFP) is not limited to operate only during a nominal 85% Turbine Runback during power operations. Therefore, the SBMFP is to be available during all phases of power operation. The FSAR requires revision (sections 10.4.7.2 "System Description" and 10.4.7.3 "Safety Evaluation" to clarify 100% guaranteed plant power with all three Feedwater (FW) Pumps operating. FSAR Change Package 1449 is being issued for the changes being made to Chapter 10.

Safety Evaluation Summary

The initial design has always provided the potential for operation of the TDMFPs and the SBMFP concurrently, however it was implied that the SBMFP was only used during certain periods of unit startup/shutdown operation, and would start automatically on the loss of a TDMFP if the Unit was above 85% guaranteed load. Although it appeared that for the new operating scenario, that the time to reach hi-hi SG level could have been less than in the current analysis of record where effects of overcooling would occur sooner, and the resulting DNBR would have been less than the results of the current analysis (1.527) yet still above the minimum acceptable (1.31), the analysis injected feedwater flow for the postulated condition bounds the actual possible injected flow, therefore there is no resulting reduction in DNBR . Accordingly the change is safe and does not represent a USQ.

SA-SE Number *WBPLMN-96-079-0*

Affected Documents	*Document Type*	*Safety Assessment Title*	*Implementation Date:*
DCN S-39001-A FSAR Change Pkg. 1456	DCN	Reclassify Boric Acid Tank Heaters	11/5/96

Description of Change, Test, or Experiments

This DCN clarifies in design output documentation that the boric acid tank (BAT) heaters (1 & 2-HTR-62-239-A & -245-B, 0-HTR-62-243-A & -246-B), are no longer required to be Class 1E devices. This change will allow the use of non-1E replacement heaters for 1-HTR-62-239-A and -245-B.

The heaters are available only as a convenient means to heat the 4 wt. % Chemical and Volume Control System (CVCS) boric acid solution if the building temperature begins to approach 63 °F. However, the heaters are not credited for maintaining the boric acid above the 58 °F solubility temperature. This is accomplished by the surrounding environmental temperature of the building itself. The normal minimum environmental temperature is specified on environmental drawing 47E235-52 to be 60 °F. In addition, the environmental temperature can drop to 50 °F for up to eight hours per excursion and will occur less than 1% of the plant life. The CVCS system description document N3-62-4001, Special Operations Section 4.1.2, addresses these temperature excursions by requiring that the boric transfer pumps be run on recirculation to add heat to the fluid if the surrounding temperature drops to within 5°F of the 4 wt. % boric acid solubility temperature of 58 °F. Based on this existing special operations requirement, the boric acid tank heaters are no longer required to Class 1E devices.

The heaters currently receive 1E electrical power. The power supply will not be changed because circuit breakers exist to protect the 1E power buses from possible faults and overloads in the heater circuits. The heaters will retain their trained designation in the component identifier number (CID). This will prevent unnecessary changes to the CID in other associated documents. However, the heater's 1E designation in the Q-List screen of EMS is changed to "N". Calculation WBN-OSG4-013 "Chemical and Volume Control System (62) NUREG-0588 Category and Operating Times" is revised to identify that the heaters are no longer classified 1E, but the heater's Category C classification will be retained to prevent any impacts to successor documents such as WBPEVAR8603004 "Class 1E Power System Category C Devices Interfaces", WBPEVAR8603005 "Failure Analysis of Category C Devices Utilized in Class 1E Power System", and WBPE0629207003 "Failure Analysis of System 62 Category C Devices". The heaters will retain their safety classification of "SR" because they are flanged into the boric acid tanks and, therefore, form part of the pressure retaining boundary of this seismic Category I component.

Installation of electrical circuit isolators is not required. Circuit breakers already exist at the taps off the 1E power busses. Since the breakers for the heaters are now used as isolation devices which protect Class 1E busses from non-qualified loads, DCAs are included in this DCN to identify the breakers on the 45A710 drawing series for periodic functional testing. This addition to the 45A710 drawing series will assure operability of the breakers in accordance with TRM 3.8.1 "Electrical Power Systems - Isolation Devices." However, no revision to TRM 3.8.1 is necessary because the individual breakers are not listed in the TRM. Instead, the breakers within the scope of TRM 3.8.1 are identified by reference to the 45A710 drawing series. In addition, the implementing Technical Requirements Instruction, O-TRI-0-2, does not require revision because the TRI simply refers to the 45A710 drawing series also.

Safety Evaluation Summary

The DCN S-39001-A changes do not affect any FSAR evaluations (accident analysis or equipment malfunctions) previously performed. The change to Table 3.2-2a does not change the system functional requirements for the CVCS as contained in section 9.3.4.2.2. No new accidents or equipment malfunctions are created. Technical Specifications, including the margin of safety as defined in the bases, are not affected. This change does not adversely affect any equipment important to safety, either directly or indirectly. Therefore, on the basis of the evaluation of effects, it is concluded that the proposed changes are acceptable from a nuclear safety standpoint and no unreviewed safety question exists.

In addition, the replacement heaters to be used in the 1-HTR-62-239-A and 2-HTR-62-245-B applications were originally purchased as non QA components and the material certifications can not be located. Therefore, a basis for acceptability to use the non-ASME heaters in the ASME Section III, Class 3 application is also provided within the scope of DCN S-39001-A.

FSAR Table 3.2-2a "Classification of Systems Having Major Design Concerns Related to a Primary Safety Function" will require notation that the replacement heaters have been evaluated and determined to be acceptable for use in the ASME Section III, Class 3 application.

SA-SE Number *WBPLMN-96-082-0*

Affected Documents	*Document Type*	*Safety Assessment Title*	*Implementation Date:*
DCN 38977-A	DCN	Condensate Polishing Demineralizer System is sampled for sodium (Na)	1/16/97

Description of Change, Test, or Experiments

The Condensate Polishing Demineralizer System (CPDS) is sampled for sodium (Na). This sample temperature has increased to approximately 120°F with the Condensate System operating temperature of 125°F - 135°F. The Na analyzer (1-NAAN-14-203) automatically bypasses the sample fluid if the temperature is > 113°F.

DCN # 38977-A adds a sample cooler to the Na analyzer. This will reduce the sample temperature and prevent the analyzer from being bypassed at high temperatures. The cooling water, just like the sample to the Na analyzer, is routed to CPDS for further processing. The CPDS is sampled and analyzed prior to release, and the release has a radiation monitor (0-RE-90-225) that automatically isolates the discharge flow.

Safety Evaluation Summary

The CPDS, its associated components, piping, and valves are located in the Turbine Building. The CPDS is normally non radioactive, non safety related, installed in a non seismic structure, and is not used during any accident. The CPDS does have the potential to be radioactive in the unlikely event of a large primary to secondary leak. However, this change ensures that the potential radioactive fluid is processed to the CPDS where the activity is verified with sampling and analysis prior to release. This DCN does not change the logic or function of any system that is important to safety. These changes are within the existing design basis limitations of the ODCM and therefore, do not represent a change to radioactive release criteria or result in higher discharge concentrations (non radioactive). A review of the detailed changes leads to the SE conclusions that this change is safe and does not constitute an USQ.

This change does not alter the system design from an operational perspective. The equipment involved in the modification does not interface with any equipment whose malfunction could result in an accident which has been evaluated in the FSAR. This DCN does not change or affect the design basis for any system that is important to safety. Additional components have been added by this change. These components, if a malfunction occurs, would not cause a radioactive releases in excess of the limits established by 10 CFR 20 and 10 CFR 100 since both the sample and cooling water are routed to the CPDS and a release from the CPDS is permitted only when the activity is below the limit as defined in ODCM. No new potential single failures of existing components will occur as a result of the new operational philosophy. Neither will this change cause this system or any system important to safety to fail to fulfill its functional requirements. The CPDS, its associated components, and piping do not perform any accident mitigation function. This change does not reduce the margin of safety identified in the applicable Technical Specifications. These changes do not prevent any component from performing its function as described in the Technical Specifications. The ODCM limits for releases from the CPDS are not revised or challenged by these changes.

SA-SE Number *WBPLMN-96-085-0*

Affected Documents	*Document Type*	*Safety Assessment Title*	*Implementation Date:*
DCN M-38965-A	Fire Protection Program	Check Valve in the Fire Protection Header	5/16/97

Description of Change, Test, or Experiments

The modification removes the internals to check valve 0-CKV-26-551. The valve has a history of leakage and frequent maintenance. The valve is not required to meet NFPA codes. The only change to the SAR is to a figure in the Fire Protection Report.

Safety Evaluation Summary

There are two A-train powered and two B-train powered electric driven fire pumps that feed the A and B train headers and combine into a common header. A check valve is provided for the outlet of each fire pump. These check valves will prevent any backflow from the train headers or common header. The train headers do not have check valves, but the original design conservatively called for a redundant check valve (0-CKV-26-551) to be installed in the common header . This valve is not required to prevent backflow through the fire pumps. There are two valves between the check valve and each trained header; therefore, it is not required for isolation of the common header from the trained headers. The check valve is not required for Appendix R compliance, Appendix A compliance, nor NFPA compliance and is not required for a design basis event; therefore, the removal of the internals of the check valve does not degrade the safety of the plant and is acceptable.

SA-SE Number *WBPLMN-96-086-0*

Affected Documents

DCN R-39046-A

Document Type

DCN

Safety Assessment Title

The line running from the high crud tanks in the condensate demineralizer system has developed leaks due to exposure to sulfuric acid errosion

Implementation Date:

11/20/96

Description of Change, Test, or Experiments

The condensate demineralizer system in the Turbine Building has developed several leaks in the discharge line to sump as a result of non-neutralized acid being discharged through this line. Ultrasonic examination (UT) revealed several thin spots in the affected carbon steel lines. UT was performed on the affected portions of the vent line, cooling tower blowdown, and various stainless steel sections. UT indicated that no degradation had occurred in these lines.

This DCN replaces the material for that portion of the system which has documented thin spots. The new material is a Nickel-Chrome steel (Carpenter 20). Carpenter 20 is more resistant to acidic corrosion than the existing carbon steel. It also has comparable strength to the existing carbon steel such that the integrity of the system is not impaired and the existing supports may be used with out redesign.

This DCN may be performed in stages.

Stage 1 includes the carbon steel piping which has leaks identified and/or is known to be thin. This includes the carbon steel piping between valves 360 and 288, between 360 and 451, between 451 and the first flange to cooling tower blow down, and between 451 and the flange at the floor to sump, associated carbon steel drains, and carbon steel sample line.

Stage 2 includes piping between valves 397 and 294 and associated carbon steel drains.

The system may be resumed to service after stage 1 has been completed. Piping identified in stage 2 is located between valves 397 and 294. Random UT indicates that excessive degradation to this piping has not occurred, therefore this piping is currently sufficient to serve its intended function.

As the piping is replaced, the carbon steel valves will be inspected visually to determine if degradation has occurred to the valve bodies and internals which would require replacement or repair. Repair and replacement to the original configuration can be performed as regular maintenance.

There are no MODE restrictions for implementation of this DCN.

Safety Evaluation Summary

The CPDS, its associated components, piping, and valves are located in the Turbine Building. The CPDS is normally non-radioactive, non-safety related, installed in a non-seismic structure. and is not used during any accident.

The CPDS does have the potential to be radioactive in the unlikely event of a large primary to secondary leak. Both sample and cooling water are routed to the CPDS to ensure that any radioactive fluid would be monitored prior to a release This DCN does not change the logic or function of any system that is important to safety. The material changes are with in the existing design basis limitations of the ODCM and therefore do not represent a change to radioactive release criteria or result in higher discharge concentrations (non-radioactive).

Carpenter 20 has a tensile strength of 91 ksi as compared to a maximum tensile strength of 70 ksi for the currently installed carbon steel piping. Carpenter 20 contains 19%-21% Chromium, 32%-35% Nickel, and 2% - 3% Moly all of which improve the resistance to degradation caused by acidic corrosion.

A review of the detailed change concludes that the material change is safe and does not constitute a unresolved safety question (USQ).

SA-SE Number *WBPLMN-96-087-0*

Affected Documents

DCN S-39016-A
FSAR Change Pkg. 1455

Document Type

DCN

Safety Assessment Title

System Description, Design Criteria, and FSAR are corrected to accurately describe the function of the Pressurizer (PZR) PORVs to provide a means to depressure the RCS following a steam generator tube rupture event.

Implementation Date:

4/2/97

Description of Change, Test, or Experiments

System description (SD) N3-68-4001 (as required by WBPER960639), Design Criteria WB-DC-40-70, and FSAR Section 5.5.13.1 are corrected to accurately describe the function of the Pressurizer (PZR) PORVs to provide a means to depressure the RCS following a Steam Generator Tube Rupture (SGTR) event. The SD and FSAR are also corrected to identify PORV use for cold overpressure mitigation (COMS). The revisions render the system descriptions, design criteria, and FSAR Chapter 5 consistent with FSAR Chapter 15, Westinghouse SGTR analysis, WB-DC-40-64, and current EOIs with respect to the functional requirements of the PZR PORVs following a SGTR event.

Safety Evaluation Summary

The changes render the affected documents consistent with the Westinghouse SGTR Analysis (Ref. 6) as well as FSAR Chapter 15 and WB-DC-40-64 (Ref. 3). Therefore these documentation only changes do not have an affect (direct or indirect) on any Plant System, Structure or Component (SSC). Additionally, since emergency operating procedures EOI E-3, ECA-3.2, and ECA-3.3 currently provide for utilization of the PZR PORVs to mitigate a SGTR event, no procedures are affected by the changes in this DCN.

SA-SE Number *WBPLMN-96-090-1*

Affected Documents	*Document Type*	*Safety Assessment Title*	*Implementation Date:*
TP-68-023 STI-96-02 CN 1,2,3	Special Test	Steam Generator Moisture Carryover Test	3/20/97

Description of Change, Test, or Experiments

A steam generator moisture carryover test was to be performed for Watts Bar Nuclear Plant (WBNP). Previous testing showed that moisture carryover average for the WBNP steam generators was 0.4% which is above the design value of 0.25%. This condition was documented in WBPER960653. High carryover has the potential to increase erosion wear in the high pressure turbine.

Moisture carryover is sensitive to at least two parameters of interest: water level; and primary system power. The primary system power impacts generator steam production and efficiency of the primary separators. The separators are also sensitive to the normal water level in the generator. WBNP uses a programmed level which changes from approximately 38% NR at 0% power to 66.5 % NR at 100% power. A lower water level of 60% NR may reduce separator drain backpressure and improve separator efficiency. In order to diagnose the carryover problem and formulate engineering solutions, additional moisture carryover tests are to be conducted. These tests will be used to determine the generator moisture carryover sensitivity to operating water level and to reactor power.

A separate phenomena of interest to WBNP is decreased steam generator longevity due to high RCS inlet temperatures to the steam generators. These high temperatures are a result of a high design TAVG temperature in the reactor coolant system. This high temperature accelerates stress corrosion cracking of the steam generator tubes and thereby shortens generator life. In order to assess this problem, a reduced TAVG test or turbine volumetric flow test is needed to determine if full turbine generator power can be maintained at a reduced RCS TAVG. Since, the reduced TAVG results in higher moisture carryover, it is desirable to measure moisture carryover during this volumetric flow test.

The testing is arranged in two separate procedures. The moisture carryover is in STI-96-02 which optionally invokes TP-68-023. Conversely TP-68-023 has been written to be run independently if moisture carryover measurement is not required. Test STI-96-02 drives the following data points:

1. 100% power, 66.5% NR SGEN LEVEL, Nominal TAVG
2. 100% power, 60.0% NR SGEN LEVEL, Nominal TAVG
3. 100% power, 60.0% NR SGEN LEVEL, Nominal TAVG -3°F, -6°F, -9°F (this test series is optional)
4. 97% power, 65.6% NR SGEN LEVEL, Nominal TAVG

Test TP-68-023 may also be conducted independent of the moisture carryover test if carryover results are not deemed necessary (i.e., moisture carryover has been performed previously and results are known). In this mode, only thermal performance data is recorded.

A radioactive sodium tracer test (Na24) will be used to measure moisture carryover.

The two tests constitute tests not described in the FSAR and were evaluated to determine if an unreviewed safety question exists. During the tests, the plant will be operated in a safe manner consistent with the Plant Technical Specifications with several systems temporarily deviating from

Safety Evaluation Summary

Conduct of the moisture carryover test and the turbine volumetric flow test do not result in significant impacts to plant transient and accident analyses described in the FSAR. Regulatory safety limits are not exceeded and the plant will not deviate from Technical Specification requirements. Two of the four test datasets are conducted with the plant operating at normal thermal hydraulic conditions (100% power and 97% power - all other parameters normal). The other two datasets require the plant steam generator level to be operated below normal (60% NR versus 66.5% NR) and for RCS TAVG to be reduced no greater than -10°F from the normal full power value of 588.2 °F. Both of these changes are within the system design capabilities as described in the FSAR. In addition, these changes have been shown not to result in initial conditions which invalidate the conclusion of the plant safety analyses and are therefore acceptable from a safety and licensing perspective.

Addition of the temporary equipment required for the radioactive tracer moisture carryover test also does not create the potential for malfunctions or transients which have not been previously analyzed since the normal plant systems will be used as the path for tracer injection. One temporary connection to the feedwater system is required in the Turbine building which will use an existing double isolated drain on the high pressure feedwater piping. This connection is between the main feedwater pumps and the high pressure feedwater heaters. Deviations from the system descriptions in the FSAR are mainly for the special protective system alignments as a result of the radioactive source tracer in the moisture carryover test and the temporary manual control of primary system TAVG for the turbine volumetric flow test. These deviations consist of temporary isolation of the steam generator blowdown system, bypass of normal condensate polishing, and alignment of the turbine building sump to the waste ponds. In addition, several radiation monitors may temporarily alarm during the test as a result of the radioactive source in the balance of plant system. These deviations will not impact the plant safety analysis.

FSAR descriptions. However, the impacts have been shown to be within the design basis and no unreviewed safety question exists.

SA-SE Number *WBPLMN-96-094-0*

Affected Documents

TACF No. 1-96-031-062 R2
SOI-68.02

Document Type

Temporary Alteration

Safety Assessment Title

Allowance of continuous leakoff flow through the reactor coolant pumps number one seal leakoff from an upper lmit of 7 gpm to 10 gpm during the plant's mid-cycle outage

Implementation Date:

10/31/96

Description of Change, Test, or Experiments

This safety evaluation concludes that it is permissible to operate either the Nos. 3 or 4 RCPs (or both) with a No. 1 seal leakoff limit of 10 gpm. With Nos. 1 and 2 pumps within normal limits the total No. 1 seal leakoff is 32 gpm. With the required 1 gpm for thermal barrier injection into the RCS and 0.5 gpm to the No. 2 seal the total required seal injection flow is approximately 38 gpm. Total seal injection flow available is 70 gpm (PDP capacity) minus 27 gpm required for the letdown heat exchanger (charging) which equals 43 gpm seal injection flow; therefore, the 38 gpm required (max) is well within the PDP capability.

Safety Evaluation Summary

The basis for the SE conclusions may be summarized as follows:

1. WCAP-11347 summarizes the RCP seal cooling for FSAR Chapter 15 accidents. This remains unimpacted.

2. WCAP-10541 Rev 2, supplements 1 and 2 summarize the station blackout requirements and these remain unimpacted. Station blackout involves the loss of all injection flow and thermal barrier cooling. It is predicated on the cavity between the thermal barrier and the No. 1 seal reaching a critical temperature. The leakoff rate being excessive lowers the time that it takes for the cavity to become filled with RCS temperature water. The WCAP considers this time frame with a open (but not failed) seal face and a corresponding leakage rate of 21 gpm. The plant IPE (Ref. 1.d) predicts an approximately 85 percent chance of the 21 gpm leakage. This leaves a 15 % chance of leakage in the range of 21 gpm to 480 gpm. The event tree sequence starts with this predicted leakage so it is evident that excessive leakoff rates for normal operation (10 gpm) do not affect SBO analysis.

3. WPT-14193 assures utilities of the adequacy of design and consistent high quality of materials furnished by Westinghouse. The cartridge seal upgrade to be implemented at midcycle assures that WBN has these designs and materials.

4. GPUs TMI-1 plant has successfully run with leakoff rates in excess of 10 gpm and the major attributes of their safety evaluation apply to WBN. Attributes applicable include two high capacity CVCS centrifugal charging pumps available for seal injection if required.

5. The WBN Safety Evaluation Report (SER) and the NRC's understanding of the plants operation remain unaffected.

SA-SE Number *WBPLMN-96-100-0*

Affected Documents	*Document Type*	*Safety Assessment Title*	*Implementation Date:*
FSAR Change Pkg. 1458	FSAR	Dose Rates Above Spent Fuel Pool	10/17/96

Description of Change, Test, or Experiments

Review of the FSAR for the spent fuel reracking effort discovered erroneous statements regarding the dose rate above the racks following the loss of water inventory in the spent fuel pool (section 9.1.3.1.2), and incorrect water depths above the spent fuel during fuel transfer (sections 9.1.4.3.4, 12.3.2.2). Additionally it was discovered that the gamma decay information found in Table 1 5A-1 did not reflect the information used by TVA codes. Also, the Table 1 5A-2 X/Q values were incorrect. WBPER960740 was written to identify and correct these errors. Details of the errors are as follows:

Section 9.1.3.1.2: States that following a loss of water inventory, a minimum of 10 feet of water will remain above the spent fuel and that the dose rate above the water is less than 2.5 mrem/hr. The 2.5 mrem/hr is correct only for a single assembly, not an entire core or spent fuel pool.

Sections 9.1.4.3.4 and 12.3.2.2: State that the minimum water shield above the active fuel during fuel transfer is 10 feet. The correct distance is 9.9 feet.

Table 1 5A-1: Nuclide decay information does not conform to the values used in TVA codes. The TVA codes use detailed information not normally presented in gamma/beta decay information.

Table 1 5A-2: The offsite 5th percentile X/Q values are out of date and need to be updated.

Page 1 5A-3: reference section needs to be updated.

This change issues FSAR change package 1458 to revise/correct the above FSAR sections.

Safety Evaluation Summary

This change involves revision of the FSAR to correct minor errors with regards to how the plant is actually configured and information used in TVA codes. These changes do not affect equipment operational requirements as currently discussed in the FSAR and as required in Technical Specifications 3.7 and 4.3 and applicable bases. Consequently, a USQ does not exist for this change because the probability or consequences of an accident or equipment malfunction is not increased by the change. In addition, the change does not create the possibility of an accident or malfunction of a different type than any previously evaluated in the SAR and the margin of safety defined in the Tech Specs bases is not reduced.

SA-SE Number *WBPLMN-96-101-0*

Affected Documents	*Document Type*	*Safety Assessment Title*	*Implementation Date:*
DCN 39064-A FSAR Change Pkg. 1423	DCN	Replace Mobile Waste Demineralization System Spent Resin Storage Containers	3/21/97

Description of Change, Test, or Experiments

The vendor supplied Mobile Waste Demineralization System (MWDS) consists of several vessels with an associated pumping skid and level control system. The vessel headers have influent and effluent isolation valves and all piping is welded with long radius bends. Demineralizer vessels are operated inside shielding in the waste packaging area with a remote control panel to insure that the dose to personnel is within acceptable limits. The system is designed to the applicable portions of Regulatory Guide 1.143, Revision 1, 1979.

The MWDS provides in-line processing of liquid radwaste through filtration and demineralization. The MWDS receives both tritiated liquid and nontritiated liquids. Processed water from the MWDS is sent to either the Chemical and Volume Control System (CVCS) Monitor Tank or the Cask Decontamination Collection Tank (CDCT) for release to the river.

The liquid radwaste is processed through ion exchange and filtration which remove soluble and suspended radioactive materials from the waste streams. The first vessel is normally loaded with a filter media, such as activated carbon, to provide initial filtration of the radwaste. This filter medium removes solids, cobalt isotopes (existing in the form of colloidal-sized suspended solids), cleaning agents, and other chemicals that can be removed by absorption of the activated carbon. A mechanical filter loaded with filter cartridges can be alternately used for filtration. This conditions the radwaste for treatment in the subsequent tanks.

The subsequent demineralizer tanks contain beds (anions and cations) of ion-exchange resins, which remove the soluble constituents of the waste stream. Once the resin and filter media is expended, the resin is removed from the MWDS vessels to the vendor supplied spent resin storage containers to accumulate enough resin for offsite disposal, and the filters are placed in a shielded container for transport and storage prior to offsite disposal.

The MWDS spent resin storage containers were supplied with manual (not remote) operated valves and are located in a High Radiation Area. During resin transfers to and from the storage containers, personnel would receive unnecessary exposure.

Design Change Notice (DCN S-39064-A revises the System Description (SD) N3-77B-4001 and N3-77C-4001 to replace the MWDS spent resin storage containers with a single spent resin disposal container. The spent resin disposal container with a High Integrity Container (HIC) placed inside. The Rad-Vault has an internal coating to aid in the prevention of the container from becoming contaminated. This container is located in the Auxiliary Building railroad bay. The MWDS spent resin will be periodically transferred to the HIC and once the HIC is full, the HIC will be dewatered, removed and placed in a shipping cask for offsite disposal.

Safety Evaluation Summary

The Waste Disposal Systems (WDSs), their associated components, piping, and valves are located in the Auxiliary, and Reactor Buildings. The WDSs are radioactive, and the MWDS spent resins have the potential to be radioactive. The WDSs are non-safety related except for containment isolation valves, which are installed in a seismic structure, and the containment isolation valves are only used during an accident. This DCN does not change the logic or function of any system that is important to safety. These systems, associated components, and piping do not perform any accident mitigation function except for containment isolation valves which have not been affected. A review of the detailed changes leads to the SE conclusions that this change is safe and does not constitute an USQ.

FSAR Section 11.2.2 identifies equipment faults which could occur with moderate frequency, including fuel cladding defects in combination with malfunction in Liquid Radwaste Processing System such as pump or valve failures or evaporator failures. Chapter 15 accident analysis identifies an unexpected and uncontrolled release of radioactive xenon and krypton fission product gases stored in a Waste Gas Decay Tank as a consequence of a failure of a single Waste Gas Decon Tank or associated piping. This DCN is not associated with the equipment that could cause these events. Also, this change is not associated with the protective features used to detect and mitigate the effects of these events. The equipment involved in this change does not interface with any equipment whose malfunction could result in an accident which has been evaluated in the FSAR.

The only equipment that could fail is the HIC itself, and this probability is considered remote since each HIC comes from the manufacturer with COC, and the HIC is inspected prior to use and before shipment offsite. In addition, the Rad-Vault concrete storage container has an internal coating to aid in the prevention of the container from becoming contaminated if any leaks from the HIC occur. The previously evaluated malfunctions of the failure of Radwaste components were reviewed and there is no increase of the consequences of these malfunctions. This change does not create the potential for a radioactive liquid effluent release pathway as defined in ODCM. No new potential single failures of existing components will occur as a result of this change. Neither will this change cause these systems or any system important to safety to fail to fulfill its components, and piping do not perform any accident mitigation function except for containment isolation which has not been affected. This DCN does not add any additional or different types of failure modes that have not been addressed in the FSAR Chapter 15. These changes do not reduce the margin of safety identified in the applicable Technical Specifications because the ODCM limits for releases from the Waste Disposal System are not revised or challenged by these changes. These changes do not prevent any component from performing its function as described in the Technical Specifications.

SA-SE Number *WBPLMN-96-106-0*

Affected Documents	*Document Type*	*Safety Assessment Title*	*Implementation Date:*
FSAR Change Pkg. 1459	FSAR	Nonintent word change to FSAR regarding manway cover doors to be closed when reactor containment integrity is required.	11/19/96

Description of Change, Test, or Experiments

This FSAR CP clarifies when the RHR sump room manway cover doors may be open for personnel access for maintenance and/or monitoring. When the ABSCE was moved from the RHR sump room to the total Auxiliary Building enclosure, the leak tightness requirements for the RHR sump room were deleted. At this point in time it became permissible for the doors to be open during any mode of operation but the FSAR did not clearly reflect this fact, however, it did state that the doors were seismically qualified in the open position. The clarification made by this FSAR CP clearly states the doors may be open during any mode of operation but are normally closed and secured in accordance with drawing 44N355 Rev E (Ref. 1.b).

Safety Evaluation Summary

The ABSCE boundary was moved in accordance with the guidance in the SRP. The criterion that are the basis for the conclusions stated in the SA/SE. These conclusions are valid for this SA/SE and their strength is that the manway cover doors do not form a primary or secondary containment boundary. This FSAR CP clarifies the wording of the FSAR text to the intent of the manway cover doors usage. Accordingly, the change is safe and does not represent a USQ.

SA-SE Number *WBPLMN-96-107-0*

Affected Documents

DCN S-39119-A
FSAR Change Pkg. 1461

Document Type

DCN

Safety Assessment Title

Use of the Containment Purge Air System while the Reactor Building is open to the Auxiliary Building in system descriptions N3-30RB-4002 and N3-30AB-4002.

Implementation Date:

3/19/97

Description of Change, Test, or Experiments

System Descriptions N3-30RB-4002 and N3-30AB-4002 contain conflicting statements regarding the operation of the Containment Purge System relative to the position of the containment hatches during modes 5 and 6. DCN S-39119-A replaces the current statements with statements such as the following which will clarify how the Unit 1 Containment Purge System is to be operated during Modes 5 and 6. The status of the containment hatches in relation to Containment Purge System operation also specified. Note that the system description revisions may not match the following statements word-for-word, but the intent will be the same.

N3-30RB-4002
The Auxiliary Building Gas Treatment System (ABGTS) is required to be operable during movement of irradiated fuel in the fuel handling area during Modes 5 and 6. At these times while the containment hatches are open, the Containment Purge System shall not be operated and 1-FCV-30-61, -62, -213, & -216 shall be closed. If the containment hatches are closed, the Containment Purge System may be operated, but 1-FCV-30-12 and -54 must be closed if the annulus is open to the Auxiliary Building.

During Modes 5 and 6 when there is NO movement of irradiated fuel in the fuel handling area, the Containment Purge System may be operated with the containment hatches open or closed.

N3-30AB-4002
During cold shutdown or refueling of Unit 1 (Modes 5 and 6), the containment and annulus may be opened to the Auxiliary Building and thus become a part of the Auxiliary Building Secondary Containment Enclosure (ABSCE) boundary. During this condition if the ABGTS is not required to be operable per Tech Specs, the Containment Purge System may be operated. An ABI signal will shut down the Purge Ventilation Fans, Instrument Room Fans and close associated isolation dampers/valves and start the ABGTS. However, as an added precaution to protect the ABGTS operational boundary, operator action is needed to ensure that the purge air system is shut down and to close 1-FCV-30-61, - 62, -213, &-216.
The Containment Purge System may not be operated (and 1-FCV-30-61, -62, -213, & -216 must be closed) if the containment hatches are open during those conditions when the ABGTS is required to be operable (I.E., during movement of irradiated fuel assemblies in the fuel handling area).

SO1-30.02 will also require revision as a result of DCN S-39119-A to ensure that the Containment Purge System is not operated and 1-FCV-30-61, 42, -213, & -216 are closed during mode 5 or 6 (with the containment hashes open) whenever irradiated fuel assemblies are being moved in the fuel handling area. The SOI will also be revised to require closure of 1-FCV-30-12 and -54 if the Containment Purge System is operated during mode 5 or 6 while irradiated fuel assemblies are being moved in the fuel handling area with containment closed, but with the annulus open to the Auxiliary Building.

SO1-30.02 currently contains Precaution and Limitation 3.N to ensure the purge air fans are stopped

Safety Evaluation Summary

The DCN S-39119-A and resulting SO1-30.02 changes do not affect any FSAR evaluations associated with the small or large break LOCA or the fuel handling events for which the ABGTS is credited for mitigation. Neither the probability of occurrence or consequences of these events is increased. The changes will assure that the ABGTS can maintain the site boundary and low population zone radiological dose rates below 10 CFR 100 limits following a postulated fuel handling accident during mode 5 or 6 with the containment hatches open. The Failure Modes and Effects Evaluation performed to demonstrate that the ABGTS can perform its accident mitigation function is unaffected. The functional requirements of the ABGTS or the Auxiliary Building Secondary Containment Enclosure are not affected. The functional requirements of the safety related portions of the Containment Purge System are also unaffected. No new accidents or equipment malfunctions are created by the measures implemented to assure that the ABSCE is maintained secured whenever irradiated fuel assemblies are being moved in the fuel handling area during modes 5 or 6 with the containment hatches open. Technical Specifications 3.6.1, 3.7.12, 3.9.4, 3.9.8, 3.3.6 associated with the operability of the ABGTS and containment, including the margin of safety as defined in the bases, are not affected. This change does not adversely affect any equipment important to safety, either directly or indirectly. Therefore, on the basis of the evaluation of effects, it is concluded that the proposed changes are acceptable from a nuclear safety standpoint and no unreviewed safety question exists.

if Auxiliary Building Isolation (ABI) is required during cold shutdown, as an added precaution to protect the ABGTS operational boundary. This precaution will have to be supplemented as a result of DCN S-39119-A to also require closure of 1-FCV-30-61, -62, -213, & -216 and to also make the precaution applicable to mode 6 (if containment is open).

FSAR sections 6.2.3.2.1.3 and 6.2.3.3.3 are revised by FSAR Change Package 1461 to clarify that the purge air ventilation system is only required to be shutdown in modes 5 and 6 (if the containment hatches are open) during movement of irradiated fuel assemblies in the fuel handling area.

SA-SE Number *WBPLMN-96-108-0*

Affected Documents	*Document Type*	*Safety Assessment Title*	*Implementation Date:*
DCN S-39123-A FSAR Change Pkg. 1460	DCN	Revision to FSAR to clarify the effects on the main condenser for the 40% load rejection.	6/30/97

Description of Change, Test, or Experiments

For the 50% load rejection test performed on or about September 27, 1996, it was observed that the main condenser vacuum in zone C peaked at 7.27 in HgA. Section 4.8 of the system description N3-2-4002 states when above 90% power level the main turbine should be manually tripped at 6 in HgA in either zone B or C. The turbine was subsequently tripped. At the start of the test, the inlet circulating water temperature was approximately 85°F and the vacuum of the condenser was about 4.5 in HgA. Based upon the design conditions for the main condenser, i.e., at 100% power with an inlet circulating water temperature of 70°F, the vacuum in zones A, B. and C should be 1.63, 2.38, and 3.40 in HgA respectively as stated in N3-2-4002, section 3.1.2 and FSAR section 10.4.1.2. FSAR section 10.4.1.2 and N3-2-4002 section 3.1.2 also states, "The condenser can accept a bypass steam flow of approximately 40% of maximum guaranteed steam generator flow, without exceeding the turbine high back pressure trip point or an exhaust hood temperature of 170°F with a circulating water temperature of up to 96°F." This statement is misleading in that the 40% steam bypass capacity is based on initial design conditions and will be less than 40% based upon a higher inlet circulating water temperature and should not state "of up to 96°F". This phenomenon is evidenced by reference to the Westinghouse "Steam Systems Design Manual", Revision 3, Figure 5-2-5. This DCN will revise N3-2-4002 and the FSAR to provide this clarification. This DCN is not a staged DCN.

Safety Evaluation Summary

A USQ does not exist for DCN S-39123-A for the condenser 40% load rejection design parameters because this design change provides only a clarification of the design conditions under which the 40% load rejection occurs. The probability or consequences of an accident or equipment malfunction is not increased by the change. In addition, the change does not create the possibility of an accident or malfunction of a different type than any previously evaluated in the SAR and the margin of safety defined in the Tech Specs is not reduced. The steam dump system is not a safety grade system required for the safe shutdown of the plant.

SA-SE Number *WBPLMN-96-112-0*

Affected Documents	*Document Type*	*Safety Assessment Title*	*Implementation Date:*
DCN R-39046-A	DCN	Waste Tank Lines in the Condensate Demineralizer System Has Developed Leaks Due to Exposure to Sulfuric Acid Corrosion	11/20/96

Description of Change, Test, or Experiments

The condensate demineralizer system in the Turbine Building has developed several leaks in the discharge line to sump as a result of non-neutralized acid being discharged through this line. Ultrasonic examination (UT) revealed several thin spots in the affected carbon steel lines. UT was performed on the affected portions of the vent line, cooling tower blowdown, and various stainless steel sections. UT indicated that no degradation had occurred in these lines.

DCN R-39046-A was initiated to replace the material for that portion of the system which has documented thin spots. The new material is a Nickel-Chrome steel (Carpenter 20). Carpenter 20 is more resistant to acidic corrosion than the existing carbon steel. It also has comparable strength to the existing carbon steel such that the integrity of the system is not impaired and the existing supports may be used with out redesign.

DCN R-39046-A may be performed in stages.

Stage 1 includes a large percentage of the carbon steel piping which has leaks identified and/or is known to be thin. This includes the carbon steel piping between valves 360 and 288, except as noted in Stage 2, between 360 and 451, between 451 and 0-ISV-14-151, and between 451 and the flange at the floor to sump, associated carbon steel drains. This stage also adds valves 0-ISV-14-151, -206, and-208.

Stage 2 is the portion of piping not installed in Stage 1, including 1) a portion of piping from 0-ISV-14-151 to the first permanent flange on the carbon steel line to CTBD, 2) the drain line to 0-DRV-14-570, 3) piping between 0-ISV-14-206 and 0-ISV-14-208, 4) piping between 0-FCCV-14-294 and 397. These portions are defined in TACF 0-96-47-14.

The system may be resumed to service after stage 1 has been completed. Piping identified in stage 2 will have a temporary hose attached to allow the system to be resumed to service and completed later. This is defined in TACF 0-96-47-14.

As the piping is replaced, the carbon steel valves will be inspected visually to determine if degradation has occurred to the valve bodies and internals which would require replacement or repair. Repair and replacement to the original configuration can be performed as regular maintenance.

This TACF is being issued to allow the system to be resumed to service before Stage 2 of DCN R-39046 is complete. This TACF will install a short section of temporary hose between O-ISV-14-151 and the first flange to the cooling tower blowdown; install a short section of hose between the new isolation valves between O-FCV-14-360 and O-FCV-14-288; install hose between 0-FCV-14-294 and 0-FCV-14-397. It also omits the drain line to O-DRV-14-570 until stage 2 of the DCN is complete.

Safety Evaluation Summary

The CPDS, its associated components, piping, and valves are located in the Turbine Building. The CPDS is normally non-radioactive, non-safety related, installed in a non-seismic structure, and is not used during any accident.

The CPDS does have the potential to be radioactive in the unlikely event of a large primary to secondary leak. Both sample and cooling water are routed to the CPDS to ensure that any radioactive fluid would be monitored prior to a release. This TACF and DCN does not change the logic or function of any system that is important to safety. The material changes are with in the existing design basis limitations of the ODCM and therefore do not represent a change to radioactive release criteria or result in higher discharge concentrations (non-radioactive).

Carpenter 20 has a tensile strength of 91 ksi as compared to a maximum tensile strength of 70 ksi for the currently installed carbon steel piping. Carpenter 20 contains 19%-21% Chromium, 32%-35% Nickel, and 2%-3% Moly all of which improve the resistance to degradation caused by acidic corrosion.

The hose is Goodyear XLPE, which is a polyethelene material with a 150 psi working pressure. The design pressure for this section of the system is 150 psi with a lower operating pressure. Polyethelene material is resistant up to and including 50% concentrations of sulfuric acid. Concentrations of sulfuric acid above 50% can reduce the yield and tensile strength values by a minimum of 20%. However, per vendor information the hose has a 4 to 1 safety factor and therefore should be more than adequate for this application.

A review of the detailed change concludes that the material change is safe and does not constitute a unresolved safety question (USQ).

The hoses will be attached by hose clamps to the stainless steel pipe nipple which is threaded into a stainless steel flange and then bolted to the temporary slip on type flange. The temporary slip on type flange will then be welded to the permanent pipe. The hoses may also be clamped to the nipples welded to the new 3" isolation valves being installed by this DCN. Installation of the three new 3" isolation valves will allow completion of the modification without system shutdown and will allow future repairs or modifications to the cooling tower blowdown line without system shutdown.

The new valves are 3" diameter 300# stainless steel, type 304, diaphragm valves. The diaphragm material is EPT which is resistant to sulfuric acid at all concentrations. The type 304 stainless steel replacement material is not as resistant as Carpenter 20, but is more resistant than the original system material (carbon steel).

The hose is Goodyear XLPE, which is a polyethelene material with a 150 psi working pressure. The design pressure for this section of the system is 150 psi with a lower operating pressure. Polyethelene material is resistant up to and including 50% concentrations of sulfuric acid. Concentrations of sulfuric acid above 50% can reduce the yield and tensile strength values by a minimum of 20%. However, the hose has a 4 to 1 safety factor and should be more than adequate

The system discharge is required by procedure to be neutralized, therefore the materials should be acceptable for temporary use.

There are no MODE restrictions for implementation of this TACF or DCN.

SA-SE Number *WBPLMN-96-123-0*

Affected Documents	*Document Type*	*Safety Assessment Title*	*Implementation Date:*
TP-77-025 R0	Temporary Procedure	Removal of a Waste Gas Compressor From Service for Maintenance	1/17/97

Description of Change, Test, or Experiments

Temporary Procedure TP-77-025 establishes the steps necessary to isolate a single waste gas compressor from service for maintenance. This procedure will require temporary closure of both 1-ISV-62-957 & 2-ISV-62-957 to facilitate removal of the spool piece in the discharge header from the relief valve and gas unloader valve on the malfunctioning waste gas compressor's moisture separator. Blind flanges can then be installed in place of the spool piece and the operational Waste Gas (WG) Compressor can be resumed to service. Temporary closure of 1-ISV-62-957 & 2-ISV-62-957 is necessary to maintain the Holdup Tank (HUT) pressure boundary while the malfunctioning WG Compressor is being isolated from the rest of the Waste Disposal System. After completion of maintenance on the affected WG Compressor, temporary closure of 1-ISV-62-957 & 2-ISV-62-957 is again required to remove the blind flanges and reinstall the spool piece. Per note 17 on 1-47W809-2 concurrent closure of these valves is not permitted. The note ensures a relief path is provided for the WG Compressor Packages' relief valves 0-RFV-77-758A & 0-RFV-77-758B and for Spent Resin Storage Tank (SRST) relief valves 0-RFV-77-700 and 0-RFV-77-3001 when the WG Compressor Packages and SRST are in service. The special requirements of this SA/SE ensure that sources of over-pressurization are eliminated for the WG Compressor Packages and the SRST while 1-ISV-62-957 & 2-ISV-62-957 are closed, thereby removing this equipment from service and eliminating the need for over-pressure protection.

The FSAR is not impacted by TP-77-025. Flow Diagram 1-47W809-2 is FSAR Figure 9.3-15, Sht 2. However, 1-47W809-2 does not require revision because the special requirements of this SA/SE eliminate the sources of over-pressurization for the system portions protected by relief valves 0-RFV-77-758A & 0-RFV-77-758B, 0-RFV-77-700, and 0-RFV-77-3001.

The Watts Bar Safety Evaluation Report (SER) and Supplements 1-20 have been evaluated for impact. TP-77-025 does not impact the NRC's understanding of the design and operation of WBN as described in the FSAR. The NRC has reviewed and granted acceptance of WBN request to lock open 1-ISV-62-957 and 2-ISV-62-957 in lieu of the ASME Code requirement to install positive controls and interlocks on these block valves contained in relief valve discharge lines. Since the special requirements of this SA/SE essentially removes the Waste Gas Compressor Packages and the SRST from service such that over-pressurization is no longer possible during implementation of TP-77-025, the conclusions reached by the NRC in their evaluation are not impacted. FSAR Table 3.2-2a "Classification of Systems Having Major Design Concerns Related to a Primary Safety Function" which documents NRC review of this subject.

Safety Evaluation Summary

New procedure TP-77-025 does not affect any FSAR Chapter 15 fault or operational transient evaluations. Nor does it increase the probability of occurrence or consequences of the analyzed event of a Waste Gas Decay Tank rupture. Neither the probability of occurrence or consequences of a radwaste processing system leakage event is increased. The possibility for an accident or malfunction of a different type than any previously evaluated in the FSAR is prevented by the special requirements to tag out electric power to the Waste Gas Compressors and isolate all incoming sources of pressurization to the SRST. No Tech Spec margins of safety are reduced. This change does not adversely affect any equipment important to safety, either directly or indirectly. Therefore, on the basis of the evaluation of effects, it is concluded that TP-77-025 is acceptable from a nuclear safety standpoint and no unreviewed safety question exists.

SA-SE Number *WBPLMN-96-126-0*

Affected Documents

DCN 39219-A

FSAR Change Pkg. 1465

Document Type

DCN

Safety Assessment Title

Sequential and Continuous O2 Analyzers

Implementation Date:

1/15/98

Description of Change, Test, or Experiments

The auxiliary services (Sampling System [43]) portion of the Gaseous Waste Process System (GWPS) consists of two automatic gas analyzers. One automatic sequential gas analyzer determines the quantity of oxygen (O2) (0-IO2N-43-232) and hydrogen (H2) (0-IH2N-43-231) in the gas space of several tanks, and provides a local and main control room (MCR) alarm on 2% oxygen concentration (Hi alarm), and 4% concentration (Hi-Hi alarm). A second oxygen monitor (0-O2AE-43-227) is installed to continuously sample the discharge of the operating gas compressor. This monitor sounds an alarm at 2% oxygen (Hi alarm) and 4% oxygen (Hi-Hi alarm) in the MCR. Operator action is relied upon to prevent the formation of a combustible gas mixture.

Design Change Notice (DCN) S-39219-A, which is a documentation change only, revises the System Description Document (SDD) N3-77A-4001. The SDD revision was required to more accurately depict, as stated above, the sequential O2 analyzer Hi and Hi-Hi alarm, where these alarms annunciate, that the alarms will reflash, and to differentiate between the sequential and continuous monitors. The SDD was also revised to delete the Explosive Gas and Storage Tank Radioactivity Monitoring Program, add liquid nitrogen (N2) supply, and to clarify H2 monitoring. The H2 concentration may be monitored by the sequential analyzer. However, the H2 concentration is assumed to exceed the lower flammability limit. Therefore, operator action for the sequential analyzer is based solely on the O2 concentration. If the H2 concentration is low (i.e. < 4%), this may be considered a mitigating factor when determining contingency actions for high or high-high O2 concentration. The second alarm (Hi-Hi) on the sequential analyzer was connected by DCN M-39216-A to provide the Hi-Hi alarm function. In addition to the SDD change, the Configuration Control Drawing (CCD) 1-47W830-1 was revised to show that manual isolation valve 1-ISV-77-593 is normally open. This valve manually isolates the Reactor Coolant Drain Tank (RCDT) from the Waste Gas vent header. Since the RCDT is normally aligned to the vent header, the valve should be normally open. The FSAR was revised to add liquid nitrogen (N2) supply, to clarify that there is not a Hydrogen Recombiner on the GWPS, and to clarify H2 monitoring.

Safety Evaluation Summary

The Waste Disposal Systems (WDSs), their associated components, piping, and valves are located in the Auxiliary, and Reactor Buildings. The WDSs are radioactive, are non-safety related except for containment isolation valves, which are installed in a Seismic Category I structure, and the containment isolation valves are only used during an accident. This DCN does not change the logic or function of any system that is important to safety. These systems, associated components, and piping do not perform any accident mitigation function except for containment isolation valves which have not been affected. A review of the detailed changes leads to the SE conclusions that this change is safe and does not constitute an USQ.

FSAR Section 11.2.2 identifies equipment faults which could occur with moderate frequency, including fuel cladding defects in combination with malfunction in Liquid Radwaste Processing System such as pump or valve failures or evaporator failures. Chapter 15 accident analysis identifies an unexpected and uncontrolled release of radioactive xenon and krypton fission product gases stored in a Waste Gas Decay Tank as a consequence of a failure of a single Waste Gas Decay Tank or associated piping. This DCN is not associated with the equipment that could cause these events. However, this DCN does more accurately depict, in the SDD, the monitoring equipment that is used to prevent an explosive gas mixture in the GWPS. Also, this change is not associated with the protective features used to detect and mitigate the effects of these events. The equipment involved in this change does not interface with any equipment whose malfunction could result in an accident which has been evaluated in the FSAR.

This documentation only DCN revises the SDD to more accurately depict the monitoring equipment that is used to prevent an explosion gas mixture in the GWPS. No change has been made to either new or existing equipment. The previously evaluated malfunctions of the failure of Radwaste components were reviewed and there is no increase of the consequences of these malfunctions. This change does not create the potential for a radioactive release in excess of the 10 CFR 20 and 10 CFR 100 limits since this DCN does not create a new radioactive liquid effluent release pathway. No new potential single failures of existing components will occur as a result of this change. Neither will this change cause these systems or any system important to safety to fail to fulfill its functional requirements. This system, associated components, and piping do not perform any accident mitigation function except for containment isolation which has not been affected. This DCN does not add any additional or different types of failure modes that have not been addressed in the FSAR. These changes do not reduce the margin of safety identified in the applicable Technical Specifications (TS) because the ODCM limits for releases from the Waste Disposal System are not revised or challenged by these changes. These changes do not prevent any component from performing its function as described in the TS.

SA-SE Number *WBPLMN-97-004-0*

Affected Documents	*Document Type*	*Safety Assessment Title*	*Implementation Date:*
DCN S-39269-A AOI 30.2	Fire Protection Program	Moisure Separator Reheater (MSR) Inlet and Bypass Isolation Valves	3/26/97

Description of Change, Test, or Experiments

The Appendix R Fire Safe Shutdown (FSSD) evaluation required the Moisture Separator Reheater (MSR) inlet and bypass isolation valves (1-FSV-75, 77, 79, 84, 91, 98, 27S,277,279,284,291 and 298) to be manually closed from the main control room for postulated fires in some Auxiliary Building locations (e.g., main steam valve rooms). The closure of these valves was required when the main steam isolation valves (MSIV) did not or could not close due to fire induced damage to the MSIV or their electrical circuits. Additional evaluations have determined that the MSR inlet isolation valves can remain open for up to 4 hours without jeopardizing the fire safe shutdown capability. This change documents that the manual action to close the MSR inlet isolation valves for the main control room also allows up to 4 hours to ensure that all 12 of the valves have closed.

Safety Evaluation Summary

The allowance of up to 4 hours to ensure isolation of the MSRs does not change the functional requirements of a safety system nor change the functional requirement for achieving fire safe shutdown. The maximum potential steam loss during a postulated Appendix R fire is bounded by the analysis for an accidental repressurization of the main steam system. It does not exceed the make up capability of the Auxiliary Feedwater system and is within the parameters of the cool down rate for an Appendix R event; therefore, this change is acceptable.

SA-SE Number *WBPLMN-97-007-0*

Affected Documents	*Document Type*	*Safety Assessment Title*	*Implementation Date:*
DCN W-39261-A	DCN	Replacement of Leeds & Northrup Oxygen 2 Analyzer	2/6/97

Description of Change, Test, or Experiments

The Continuous Gas Analyzer (0-02AN-43-227) automatically samples the discharge of the running gas compressor. The Oxygen 2 (O2) analyzer is used to determine the volume percent of O2 in the gas stream as the waste gases are being transferred to the in-service Waste Gas Decay Tank. This analyzer alarms in the main control room (MCR) and requires operator action to prevent an explosive gas mixture. This O2 analyzer is a Tech Spec requirement and is required for all modes of operation. If the analyzer is not operating for more than 30 days a report, is required to be filed with the NRC. The instrument could not be calibrated, and after review of the design requirements of the instrument it was determined that the instrument would not function properly in the current application, is obsolete, and is sensitive to minor changes in flow and differential pressure. Therefore, the instrument is required to be replaced.

Design Change Notice (DCN) NO. 39261-A replaces the Leeds & Northrup O2 analyzer with an analyzer from Orbisphere Laboratories. The new O2 analyzer 1S a microprocessor controlled panel mount indicating instrument which has integral software that can be verified by normal calibration processes. This new instrument is accurate to pressure changes (within the range of O - 6 bar) and flow changes (within the range of 100 - 3,000 cc/min). A flow indicating controller is being added to maintain the flow within a desired range (100 - 1,000 cc/min).

Safety Evaluation Summary

The Continuous O2 analyzer, its associated components, piping, and valves are located in the Auxiliary Building general area on elevation 713. This analyzer does not perform any primary safety function, is installed in a seismic structure, and is not used during any accident. The Chapter 15 accident analysis identifies an unexpected and uncontrolled release of radioactive xenon and krypton fission product gases stored in a Waste Gas Decay Tank as a consequence of a failure of a single Waste Gas Decay Tank or associated piping. This modification will not increase the consequences of an accident previously evaluated, and is bounded by the existing analysis of the Waste Gas Decay Tank. This DCN does not change the logic or function of any system that is important to safety. A review of the detailed changes leads to the SE conclusions that this change is safe and does not constitute an USQ.

FSAR Section 11.2.4 identifies equipment faults which could occur with moderate frequency, including fuel cladding defects in combination with malfunctions in the Liquid Radwaste Processing System such as pump or valve failures or evaporator failures. This DCN is not associated with the equipment that could cause these events. Also, this change is not associated with the protective features used to detect and mitigate the effects of these events. The equipment involved in the modification does not interface with any equipment whose malfunction could result in an accident which has been evaluated in the FSAR. This DCN does not change or affect the design basis for any system that is important to safety.

No new potential single failures of existing components will occur as a result of replacing the Continuous O2 analyzer. Neither will this change cause this system or any system important to safety to fail to fulfill its functional requirements. The Continuous O2 analyzer, its associated components, and piping do not perform any accident mitigation function. These changes do not affect any equipment required for safe operation or shutdown. In the event of a DBE, all safety related equipment is expected to operate as designed to limit the consequences of the DBE.

These changes do not reduce the margin of safety identified in the applicable Technical Specifications. These changes do not prevent any component from performing its function as described in the Technical Specifications. The ODCM limits for releases from the Waste Gas Disposal System are not revised or challenged by these changes.

SA-SE Number *WBPLMN-97-008-0*

Affected Documents	*Document Type*	*Safety Assessment Title*	*Implementation Date:*
DCN M-39190-A FSAR Change Pkg. 1468	DCN	Service Building Sewage Ejector Pumps	7/12/97

Description of Change, Test, or Experiments

The 30 gallon (60 gallons total) air operated sewage ejector pumps (0-PMP-40-EJSE and 0-PMP-40-EJSW) that process sewage for the Service Building (SB) and Control Building (CB) have failed several times. When these pumps fail, raw sewage backs up in the SB machine shop area and bathrooms.

Design Change Notice (DCN) No. 39190-A removes the existing Air Operated sewage ejector pumps and associated piping and pipe supports and replaces them with electrically operated duplex sewage grinder pumps. These pumps are located inside a 36" 0 X 6' high fiberglass basin which contains level switches whose function is to control basin fluid level. The existing pumps are not reliable and are frequently out of service. The electrically operated pumps are adequate to perform the required function, are conveniently repaired and are approximately 1/3 the cost of replacing existing pumps.

Safety Evaluation Summary

The SB sewage system, its associated components, piping, and valves are located in the Service Building (SB). This sewage is normally non radioactive, non safety related, installed in a non seismic structure, and is not used during any accident. This DCN does not change the logic or function of any system that is important to safety. A review of the detailed changes leads to the SE conclusions that this change is safe and does not constitute an USQ.

This change does not alter the system design from an operational perspective. The equipment involved in the modification does not interface with any equipment whose malfunction could result in an accident which has been evaluated in the FSAR. This DCN does not change or affect the design basis for any system that is important to safety. Additional components have been added by this change. These new components, if a malfunction occurs, would not cause a radioactive releases in excess of the limits established by 10 CFR 20 and 10 CFR 100 since the sewage system is non radioactive. No new potential single failures of existing components will occur as a result of the new electrical sewage grinder pumps. Neither will this change cause this system or any system important to safety to fail to fulfill its functional requirements. The SB sewage system, its associated components, and piping do not perform any accident mitigation function. The SB sewage system is non safety related and is not the subject of any Technical Specification requirements. This change does not reduce the margin of safety identified in the applicable Technical Specifications. These changes do not prevent any component from performing its function as described in the Technical Specifications.

SA-SE Number *WBPLMN-97-009-1*

Affected Documents	*Document Type*	*Safety Assessment Title*	*Implementation Date:*
DCN M-38747-A & F-39478-A FSAR Change Pkg. 1485	DCN	Moisture Separator/Reheater (MSR) Package	8/1/97

Description of Change, Test, or Experiments

DCN M-38747-A implements modifications that will improve the performance of the Moisture Separator/Reheater packages and enhance the operation of the equipment as described below.

EQUIPMENT DESCRIPTION

The Moisture Separator/Reheaters (MSRs) are part of the turbogenerator package supplied to WBN by Westinghouse. There are six (6) parallel MSRs located in the cycle steam path between the high pressure turbine and the three (3) low pressure turbines. There are 2 MSRs for each low pressure turbine. They receive cycle steam that is discharged from the high pressure turbine, provide moisture separation to remove condensate that has condensed in the steam, and then reheat the steam with the 1st (Low Pressure) and 2nd (High Pressure) Stage Reheaters, which are included in the equipment package. The cycle steam then enters the Low Pressure turbines.

SUMMARY DESCRIPTION OF THE MODIFICATION

The modifications being implemented to improve the operation and performance of the Moisture Separator/Reheaters (MSRs) involve the following activities:

Upgrade of the MSRs themselves,

Upgrade of existing instrumentation and addition of new instrumentation, and

Modification of the operating (steam scavenging) vent lines on the High Pressure (HP) and Low Pressure (LP) Reheaters.

MODIFICATION OF THE MOISTURE SEPARATOR/REHEATERS (MSRs)

DCN M-38747-A is modifying the 6 Unit I MSRs by replacing the existing HP and LP Reheaters, and upgrading the Moisture Separator (MS), steam distribution manifolds and internal baffles in the MSRs to improve the performance of the MSRs. Upgrading the MSs involves in part; installing bellmouths at the inlet to the steam manifolds, modifying the existing steam manifolds to have the steam discharge from the top of the manifold instead of the bottom of the manifold, installing perforated flow distributor plates upstream of the chevron vanes to improve steam flow through the chevron vanes, replacing the existing chevron vanes in the MSs with new chevron vanes, and installing flow converters and skimmer plates to improve steam flow inside the MSRs and to minimize the re-entrainment of condensate into the steam.

HP AND LP REHEATER OPERATING VENT LINES MODIFICATION

The existing Low Pressure (1st Stage) Reheater 4th pass operating vents (steam scavenging vents) lines' fixed control orifice will be replaced with a variable control orifice (globe valve). This will ensure optimum performance of the Reheater bundles when used in conjunction with the thermocouples being installed on the 4th pass tube sheet of the Reheaters. The existing High Pressure (2nd Stage)

Safety Evaluation Summary

The Moisture Separator/Reheaters are a part of the turbogenerator system. They are located in the cycle steam path between discharge from the high pressure turbine and the inlets to the three (3) low pressure turbines. The High and Low Pressure Reheaters receive heating steam from the non-safety related portion of the Main Steam System and the Number I Extraction Steam systems respectively. The heating fluid discharged from the Reheaters enters the Number 1 and 2 feedwater heater groups respectively. The Condensate discharge from the Moisture Separator in the MSRs discharges into the Number 3 Heater Drain Tank in the Heater Drains and Vents System. None of these systems are safety related, and with the exception of the non-safety related portion of the Main Steam System, they do not interface with any safety related systems. All the affected equipment is located in the Turbine Building. The electrical equipment being modified serves only non-safety related indication functions. The equipment has no control functions, can not shutdown the Turbines, and will not cause a reactor trip.

The modifications to the Main Steam System do not affect any portion or function of the safety related portion of the Main Steam System.

These modifications do not increase the probability of occurrence or the consequences of any accident previously evaluated in the SAR. The probability of a steam generator tube rupture due to copper contamination will be decreased by the replacement of the High and Low Pressure Reheater copper tubes with tube made of stainless steel.

These modifications do not increase the probability of occurrence or the consequences of any malfunction previously evaluated in the SAR, since no equipment is being installed which would adversely affect the performance of any existing safety related or non-safety related equipment.

These modifications do not increase the probability of occurrence or the consequences of an accident of a different type than any previously evaluated in the SAR, since these modifications are improvements to existing equipment and systems and do not significantly change the operation of the system or add equipment of a different type which would contribute to an accident.

These components are not included in the Tech Specs, and therefore do not have any effect on the margin of safety as defined in the basis for any Technical Specification.

Reheater 4th pass operating vents (steam scavenging vent condenser) lines' fixed control orifice will be replaced with a straight section of pipe, since calculations have shown that the existing line size is too small to pass either the current or the new design flow. The design flow for both the LP and HP Reheaters 4th pass operating vents is basically the same before and after these modifications are implemented. The performance of this equipment can be evaluated to determine if increasing the line size can be justified by a corresponding increase in performance of the MSRs.

The operating vent lines' connections to the HP and LP Reheater channel head drains are also being changed due to the improved design of the Reheaters.

INSTRUMENTATION EVALUATION

The existing flow, pressure, and temperature instrumentation in the Main Steam. Extraction Steam, and Heater Drains and Vents piping which supports the operation of the MSRs has been evaluated. As a result of the evaluations, the following instrumentation changes are being made.

UPGRADE OF THE MAIN AND EXTRACTION STEAM SUPPLY LINES FLOW ELEMENTS, HIGH PRESSURE AND LOW PRESSURE DRAIN TANK DRAIN LINE FLOW ELEMENTS; AND ADDITION OF MSRs' BELLY DRAIN TANK DRAIN LINE FLOW ELEMENTS

DCN M-38747-A replaces existing flow instrumentation (annubars and transmitters) in the Main Steam and Extraction Steam piping to the MSRs.

It also replaces the existing flow elements (orifice plates) in the drain piping from the HP and LP Reheater Drain Tanks. The existing orifice plates are being replaced with orifice plates which are better suited to the system operating conditions. The associated instrumentation is not being changed.

The MSR Belly Drain Tanks do not currently have any flow instrumentation in their drain lines. Orifice plates are being installed in these lines to permit flow measurement. The instrumentation for the MSR drain line flow elements is similar to that supplied for the High Pressure and Low Pressure Reheater Drain Tank drain lines.

PRESSURE AND TEMPERATURE INSTRUMENTATION OF THE MSRs

DCN M-38747-A also adds pressure test instrument connections and thermocouples to the MSRs' HP and LP Reheaters and to the MSR Shell internals. The instrumentation changes are being made to better determine the performance of the MSRs and to facilitate monitoring and controlling the operation of the MSRs.

SAR IMPACT

The SAR will be revised to include several revised figures which will reflect the changes made to the flow diagrams, control diagrams, logic diagrams, and heat balance diagrams as a result of these modifications. Minor changes are required to the text for clarifying the reference to the heat balance drawings. The WBN heat balance was recalculated to address the MSR performance and results in revised heat balance drawings. FSAR Change Request 1485 is prepared to make the text changes and the associated figure changes related to the heat balance. Other SAR figure changes are not required to be in the change package, since they will be updated by the normal amendment process.

SA-SE Number *WBPLMN-97-010-1*

Affected Documents	*Document Type*	*Safety Assessment Title*	*Implementation Date:*
DCN W-39512-A	DCN	TACF Replaces Flow Restriction Capillaries in the Inlet Tubing to both Oxygen and Hydrogen Analyzers with Needle Valves	9/15/97

Description of Change, Test, or Experiments

The auxiliary services (Sampling System (43)) portion of the Gaseous Waste Process System (GWPS) has an automatic sequential gas analyzer. This analyzer determines the quantity of oxygen (O2) (0-I02N-43-232) and hydrogen (H2) (O-IH2N-43-231) in the gas space of several tanks and provides a local and main control room (MCR) alarm on 2% oxygen concentration (Hi alarm) and 4% concentration (Hi-Hi alarm). This analyzer is used for all modes of operation as part of WBN's program for explosive gas monitoring, which is required by Technical Specification (TS) Section 5.7.2.15. If either alarm annunciates, operator action is relied upon to prevent an explosive gas mixture. The sequential O2 and H2 instruments have had difficulty in calibration due to inadequate flow. Flow restriction capillaries, which limit flow to 90 - 110 cc/min for both instruments (0-I02N-43-232 and O-IH2N-43-231), have been replaced several times. When new restrictors are initially reinstalled, the system operates adequately. The flow restriction capillaries should be replaced with a device that will provide equivalent flow control, but be less susceptible to clogging.

Temporary Alteration Control Form (TACF) # 0-97-004-043 replaces the flow restriction capillaries in the inlet tubing to the O2 (0-I02N-43-232) and H2 (0-IH2N-43-231) analyzers. The capillaries are being replaced with high precision needle valves. The TACF will allow this alternate design to demonstrate that the capillaries were the root cause of the flow problems and provide additional data for a permanent design change.

Design Change Notice (DCN) No. 39512-A is making TACF 0-97-004-043 that is described above permanent.

Safety Evaluation Summary

The Sequential Oxygen2 Analyzer and its associated components, piping, and valves are located in the Auxiliary Building general area on elevation 713. This analyzer does not perform any primary safety function, is installed in a seismic structure, and is not used during any accident. The Chapter 15 accident analysis identifies an unexpected and uncontrolled release of radioactive xenon and krypton fission product gases stored in a Waste Gas Decay Tank as a consequence of a failure of a single Waste Gas Decay Tank or associated piping. This change will not increase the consequences of an accident previously evaluated, and is bounded by the existing analysis of the Waste Gas Decay Tank rupture. This TACF does not change the logic or function of any system that is important to safety. A review of the detailed changes leads to the SE conclusions that this change is safe and does not constitute an USQ.

FSAR Section 11.2.4 identifies equipment faults which could occur with moderate frequency, including fuel cladding defects in combination with malfunctions in the Liquid Radwaste Processing System such as pump or valve failures or evaporator failures. This TACF is not associated with the equipment that could cause these events. Also, this change is not associated with the protective features used to detect and mitigate the effects of these events. The equipment involved in the modification does not interface with any equipment whose malfunction could result in an accident which has been evaluated in the FSAR. This TACF does not change or affect the design basis for any system that is important to safety.

No new potential single failures of existing components will occur as a result of replacing the flow restriction capillaries in the Sequential O2 Analyzer. Neither will this change cause this system or any system important to safety to fail to fulfill its functional requirements. The Sequential O2 Analyzer and its associated components and piping do not perform any accident mitigation function. These changes do not affect any equipment required for safe operation or shutdown. In the event of a DBE, all safety related equipment is expected to operate as designed to limit the consequences of the DBE. These changes do not reduce the margin of safety identified in the applicable Technical Specifications. These changes do not prevent any component from performing its function as described in the Technical Specifications. The ODCM limits for releases from the Waste Gas Disposal System are not revised or challenged by these changes.

SA-SE Number *WBPLMN-97-014-0*

Affected Documents	*Document Type*	*Safety Assessment Title*	*Implementation Date:*
DCN M-39305-A FSAR Change Pkg. 1470	DCN	Total Blowdown Flow Increased through Steam Generators.	4/22/97

Description of Change, Test, or Experiments

During startup and normal operation of the plant, the response time is not adequate to maintain the secondary side cleanup and chemistry specifications. The Steam Generator Blowdown (SGBD) flow should provide immediate response to any abnormal conditions on the secondary side to minimize the effects of corrosion within the steam generators (SGs). The SGBD flow rate should be increased, as much as possible, to ensure that the secondary side cleanup and chemistry specifications response time is adequate since the SGBD flow may be processed directly to the Condensate Polishing Demineralizer System (14) for cleanup without any dilution. Design Change Notice (DCN) No. 39305-A revises the System Description Document (SDD) N3-15-4002 to increase the total blowdown flow through four Steam Generators (SGs). The SDD was revised to make a clarification that the maximum combined blowdown flow for four SGs is 350 gpm which corresponds to approximately 87.5 gpm per SG. The existing normal operating combined blowdown flow of approximately 262 gpm which corresponds to approximately 65.5 gpm per SG was not modified by this DCN. The increased flow does not affect the SGBD radiation monitor (1-RE-90-120) since the combined blowdown flow has not been increased to the Cooling Tower Blowdown (CTBD) and flow to the CTBD would still be isolated, as required, if the stream contained high activity. The only instrumentation affected by this modification are the electronic flow transmitters (1-FT-1-152, -156, -160-, & -164) that measure the steam generator blowdown flow from each steam generator and provide input to the P2500 plant computer. The computer log points for the flow loops are connected to Emergency Response Facilities Data System (ERFDS) by data link. These flow measurements are used as input to the plant calorimetric program in accordance with the plant Technical Specifications, and the instrumentation providing these measurements are designated as "compliance instruments." This instrumentation performs no safety function.

The flow transmitters referenced above currently have a range of O - 90 GPM. Since the increased blowdown flow will be approximately 87.5 GPM per steam generator, the range must be increased to allow margin above the expected maximum flow. Accordingly, the range for these flow loops will be increased to 0 - 120 GPM. The transmitters currently installed in these loops are capable of accommodating the expanded range A software change to the plant computer (P2500) and ERFDS is required in order to change the range in these systems.

In addition to the above, the SDD and FSAR were revised to make other documentation only changes including, (1) the value for the containment pressure high setpoint was deleted since this information is contained in Setpoint and Scaling Document (SSD) SSD-1-P-30-42, and (2) the SDD N3-14-4002 was also revised to clarify that the Condensate Demineralizer Service Vessel (CDSV) pre-service rinse is normally routed to the hotwell prior to placing the vessel in service but may be routed to other locations such as to the High Crud Tanks. Configuration Control Drawings (CCDs) 1-47W801-2 and 1-47W611-1-3 were revised to correct which valves are Blowdown Isolation and which are Containment Isolation.

Safety Evaluation Summary

The SGBD and its associated components, piping, and valves are located in the Reactor Building, Auxiliary Building, and Turbine Building. The CPDS is located in the Turbine Building. The SGBD equipment does perform a primary safety function, portions are installed in a Seismic Category I structure, and is used during an accident. The CPDS, its associated components, piping, and valves are located in the Turbine Building. The CPDS is non safety related, installed in a non seismic structure, and is not used during any accident. These systems do have the potential to be radioactive in the unlikely event of a large primary to secondary leak. However, this change does not alter how potential radioactive fluid is processed to and released from CPDS. This DCN does not change the logic or function of any system that is important to safety. These changes will not increase the consequences of an accident previously evaluated. A review of the detailed changes leads to the SE conclusions that this change is safe and does not constitute an USQ.

Mainly, this DCN is associated with increasing the combined blowdown flow for the SGBD and permitting the CDSV to be routed to other location such as to the High Crud Tanks. These changes do not alter the system design from an operational perspective. This DCN does not change or affect the design basis for any system that is important to safety.

These changes do not effect any equipment required for safe operation or shutdown. In the event of a DBE, all safety related equipment is expected to operate as designed to limit the consequences of the DBE. The SGBD and CPDS are normally non radioactive but do have the potential to be radioactive after a large primary to secondary leak. No additional components have been added by this change. This change does not result in radioactive releases in excess of those established by 10 CFR 20 and 10 CFR 100 since this DCN does not create a new radioactive liquid or gaseous effluent release pathway as defined in the ODCM.

These changes do not reduce the margin of safety identified in the Technical Specifications. These changes do not prevent any component from performing its function as described in the Technical Specifications. The ODCM limits for releases from these systems are not revised or challenged by these changes.

SA-SE Number *WBPLMN-97-026-1*

Affected Documents	*Document Type*	*Safety Assessment Title*	*Implementation Date:*
DCN S-39370-B FSAR Change Pkg. 1472	DCN	Correct Valve Position for Containment Isolation Valves	6/25/97

Description of Change, Test, or Experiments

FSAR Table 6.2.4-1 is corrected to depict the normally closed position for the following valves: a) On sheet 9, 1-FCV-1-16 is currently shown in the sketch and specified in the "Valve Status" column as being normally open. This is contrary to 1-47W801-1 which shows 1-FCV-1-16 normally closed and Checklist 2 of SOI-1.01 which specifies a closed position for the valve. b) On sheet 28, 1-FCV-77-20 is currently shown in the sketch and specified in the "Valve Status" column as being normally open. This is contrary to 1-47W830-1 which shows 1-FCV-77-20 normally closed and Checklist 2 of SOI-77.04 which specifies a closed position for the valve. c) On sheet 23, 1-FCV-63-64 is currently shown in the sketch and specified in the "Valve Status" column as being normally open. This is contrary to 1-47W830-1 which shows 1-FCV-63-64 normally closed and Checklist 3 of SOI-63.01 which specifies a closed position for the valve. In addition, the "Valve Status" column of Table 6.2.4-1 will be revised to show that 1-FCV-1-16 is closed during normal operations and to show that the valve position varies for 1-FCV-77-20 and 1-FCV-63-64 during normal operations. These changes will be implemented by FSAR Change Package 1472.

The Watts Bar Safety Evaluation Report (SER) and Supplements 1-20 have been evaluated for impacts. The change to FSAR Table 6.2.4-1 does not impact the NRC's understanding of the design and operation of Watts Bar Nuclear Plant as described in the FSAR.

Safety Evaluation Summary

The changes of FSAR Change Package 1472 do not affect any FSAR Chapter 15 fault or operational transient evaluations. The design and licensing basis for the Waste Disposal, Main Steam, and Safety Injection Systems is unchanged. The normal position of valves 1-FCV-77-20, 1-FCV-1-16, and 1-FCV-63-64 has not been changed such that the function these valves perform in support of the Containment Isolation System is also not affected. Only the position shown in the sketches of FSAR Table 6.2.4-1 has been changed to agree with the design basis for the subject valves. The probability of occurrence of any of the previously analyzed events and equipment malfunctions is not increased because the design and licensing basis normal position for the subject valves has not been changed. The consequences of any of the previously analyzed events and equipment malfunctions is not increased because the "post-accident" and "power failure" position for these valves has not been changed. The possibility for an accident or malfunction of a different type than any previously evaluated in the FSAR is not increased because the manner in which the individual valves are operated is not changed and the manner in which they are operated in support of their respective system's function is also unchanged. No Tech Spec margins of safety are reduced. This change does not affect the operation of any equipment important to safety, either directly or indirectly. Therefore, on the basis of the evaluation of effects, it is concluded that FSAR Change Package 1472 is acceptable from a nuclear safety standpoint and no unreviewed safely question exists.

SA-SE Number *WBPLMN-97-027-0*

Affected Documents	*Document Type*	*Safety Assessment Title*	*Implementation Date:*
DCN M-39254-A	DCN	Cask Decontamination Collector Tank Discharge Pumps Replaced	5/7/97

Description of Change, Test, or Experiments

Cask Decontamination Collector Tank (CDCT) is a part of the Liquid Radwaste Processing System (77), and is periodically filled from the Mobile Demineralizer and then released to the environs. The CDCT discharge pumps (O-PMP-77-141 and 0-PMP-77-142) were purchased under contract 93NNB-75555C and are Goulds model 3196 STX. These pumps are normally used to transfer the CDCT to the Cooling Tower Blowdown (CTB). These pumps were originally specified to supply 100 gallons per minute (gpm) at 220 feet of head. These pumps' inboard motor bearings are overheating when either pump is operated for extended periods which causes the 10 Hp pump motors (O-MTR-77-141 and 0-MTR-77-142) to thermally overload.

Design Change Notice (DCN) NO. 39254-A replaces both pumps. The only difference is that the 10 Hp motors is being replaced with a 20 Hp and the pump impeller was increased from 7 3/8 inch to 7 3/4 inch. As a result of increasing Hp for the motors, the System Description Document (SDD) N3-77C-4001 and Configuration Control Diagram (CCD) 1-47W830-2 are being revised to delete a caution not to operate the existing pumps above the rated flow which will maintain the electrical current below the nameplate rating.

Safety Evaluation Summary

The CDCT pumps, associated components, piping, and valves are located in the Auxiliary Building general area on elevation 692. These pumps do not perform a primary safety function, are installed in a seismic structure, and are not used during any accident. The Chapter 15 accident analysis identifies an unexpected and uncontrolled release of radioactive xenon and krypton fission product gases stored in a Waste Gas Decay Tank as a consequence of a failure of a single Waste Gas Decay Tank or associated piping. This modification will not increase the consequences of an accident previously evaluated, and is bounded by the existing analysis of the Waste Gas Decay Tank. This DCN does not change the logic or function of any system that is important to safety. A review of the detailed changes leads to the SE conclusions that this change is safe and does not constitute an USQ.

FSAR Section 11.2.4 identifies equipment faults which could occur with moderate frequency, including fuel cladding defects in combination with malfunctions in the Liquid Radwaste Processing System such as pump or valve failures or evaporator failures. This DCN is not associated with the equipment that could cause these events. Also, this change is not associated with the protective features used to detect and mitigate the effects of these events. The equipment involved in the modification does not interface with any equipment whose malfunction could result in an accident which has been evaluated in the FSAR. This DCN does not change or affect the design basis for any system that is important to safety.

No new potential single failures of existing components will occur as a result of replacing the CDCT pump motors and impellers. Neither will this change cause this system or any system important to safety to fail to fulfill its functional requirements. The CDCT pumps, its associated components, and piping do not perform any accident mitigation function. The accidents and/or malfunctions associated with the Radwaste system is a failure of Waste Gas Decay Tank or associated piping and failure of radwaste components. Although this change does affect radwaste components, this equipment is not used in the mitigation of these accident/malfunctions and does not change the radiological consequences of an accident previously evaluated in the FSAR. These changes do not affect any equipment required for safe operation or shutdown. In the event of a DBA, all safety related equipment is expected to operate as designed to limit the consequences of the DBA.

These changes do not reduce the margin of safety identified in the applicable Technical Specifications. The CDCT pumps are not addressed in the Technical Specifications and these changes do not prevent any component from performing its function as described in the Technical Specifications. The ODCM limits for releases from the Liquid Radwaste Processing System are not revised or challenged by these changes.

SA-SE Number *WBPLMN-97-028-4*

Affected Documents	*Document Type*	*Safety Assessment Title*	*Implementation Date:*
DCNs S-38854-B, S-39635-A FSAR Chg. Pkg. 1488, TS Bases 3.5.2, Rev. 14 TS/TRM 97-007, (TRM 3.1.5 & 3.1.6) Rev 9	DCN	Cycle 2 Core Designs and Future Fuel Cycles	10/10/97

Description of Change, Test, or Experiments

Boron Concentration Chances for the RWST and CLAs

To accommodate Cycle 2 and future core designs, the boron concentration limits for the water stored in the refueling water storage tank (RWST) and the cold leg accumulators (CLAs) are increased to ensure that following a Large Break Loss of Coolant Accident (LBLOCA) the core remains subcritical due to the concentration of boron in the water provided by the ECCS, which takes suction from the ROOST and containment sump. The change in boron concentration limits for the ROOST has been factored into calculations to determine the maximum time to hot leg switchover to prevent boron precipitation/accumulation in bottom of the reactor vessel during recirculation and the minimum volume of berated water in the ROOST necessary to maintain shutdown margin during plant cooldown. The change in the maximum time to hot leg switchover has been updated in design output documents for incorporation in plant operating procedures.

Source Term Changes

The source term used in radiation dose calculations which support the Cycle 1 core design (the current design basis) is based on a average core burnup of 650 effective full power days (EFPDs). To support higher burnup of the Cycle 2 core design, a source term based on an average core burnup of 1000 EFPDs (and a peak assembly burnup of 1500 EFPDs) is required.

Radiation Monitor Demonstrated Range and Accuracy

The Cycle 2 source term has been used in demonstrated range and accuracy calculations for plant radiation monitors. These monitors have been shown to have acceptable range and accuracy to meet the requirements of NUREG-0737.

NUREG-0737 Mission Doses

The Cycle 2 source term has been used to determine dose to plant personnel performing post-accident (NUREG-0737) missions. The revised calculations demonstrate that the dose to plant personnel are within the limits specified in GDC 19 of Appendix A of 10CFR50.

Control Room Operator and Off-site Doses

The Cycle 2 source term has been used, as applicable, in the calculation of doses to the control room operators and off-site doses for all FSAR Chapter 15 events with consequences. The doses to control room operators and the off-site doses are within the limits specified in GDC 19 of Appendix A of 10CFR50 and 10CFR100, respectively.

The source term used in the dose calculations for the fuel handling accident is based on a maximum EOL (assembly) burnup of 1500 EFPDs. In order to maintain control room operator doses below the limits specified in GDC 19 of Appendix A of 10CFR50, the time from the occurrence of a radiation

Safety Evaluation Summary

Boron Concentration Changes for the RWST and CLAs

The purpose of the boron in the water of the RWST and CLAs is for reactivity control. The increase in boron concentration limits have been factored into the plant accident analyses and evaluated for it effects on plant equipment. The increase in boron concentration limits do not introduce a change to the function or purpose of the boron in the water stored in the RWST or CLAs. Therefore, the change in the boron concentration limits do not constitute an USQ.

[The increases in the boron concentration limits for the water stored in the RWST and CLAs require a change in the Technical Specifications (TS). These changes have been evaluated and determined to be safe, not constitute a unreviewed safety question (USQ) and have been submitted to the NRC for approval (Technical Specification Change No. 96 013).]

Source Term Chances

The increase in the average core burnup for Cycle 2 and future core designs, changes the isotopic mix and quantity of fission products contained in the fuel. This results in changes in the radiation dose to:

· plant personnel performing post-accident missions
· the control room operators
· safety-related electrical equipment and mechanical components
· off-site releases

for all accidents which result in damage to the fuel. These changes in radiation dose (or source term, as applicable) have been factored into all design basis, analyses and evaluations. Minor plant modifications have been implemented to ensure these analyses and evaluations (see A.1). The revised, design basis analyses and evaluations demonstrate that for the increased core burnup of Cycle 2 and future fuel cycles:

· structures, systems and components are able to perform their intended functions,
· the radiation dose to plant personnel performing post-accident missions and the radiation dose to the control room operators are below the limits of GDC 19 of Appendix A of 10CFR50 and
· off-site radiation doses are below the limits of 10CFR100.

Therefore, the increase in core burnup for Cycle 2 and future fuel cycles do not constitute an USQ.

Steam Line Break Outside Containment

The increase in mass and energy release for the MSLB outside containment, as a result of the Cycle 2 core design, and the resulting increase in environmental temperatures in the MSVVs has been factored into revisions to design basis analyses and evaluations. The revised, design basis level in excess of the setpoint for radiation monitors 0-RE-090-102 & -103 to closure of HVAC exhaust isolation dampers in the spent fuel pool area, 0-FCO-030-137-A, -JAMB, -140-A &-141-B, have been reduced.

The source term for the dose calculations for the steam generator tube rupture event is based on technical specification limits on RCS coolant activity during normal operation which have not changed from the Cycle 1 values. However, the activity release through the faulted steam generator has changed as a result of the Cycle 2 core design and other changes (see DCN W-39293-A) which has resulted in revision of doses to the control room operators and off-site doses for this event. The doses to control room operators and the off-site doses are within the limits specified in GDC 19 of Appendix A of 10CFR50 and 10CFR100, respectively.

Environmental Qualification

The Cycle 2 source term has been used in radiation dose calculations. The resulting doses have been used to (re)establish environmental qualification of all safety-related electrical equipment and mechanical components for the Cycle 2 core design. The changes in dose to seal material in the Residual Heat Removal System and to valve packing materials used in essential raw cooling water containment isolation valves have resulted in replacement of these materials (DCN W-39404A). All other safety-related electrical and mechanical components have been found to remain environmentally qualified.

Demonstrated Range and Accuracy of Plant Instrumentation

The Cycle 2 source term has been used in radiation dose calculations. The resulting doses have been factored into the determination of the demonstrated range and accuracy of plant instrumentation and have resulted in a change in the calibration factors used for the shield building vent radiation monitors: 1,2-RE-090-400 and the condenser vacuum exhaust accident range noble gas monitor, 1-RE-090-404 (see DCN W-39536-A). The change in source term has resulted in changes to maximum acceptable background to 0-RE-090-125, -126, -205 & - 206. The demonstrated range of all other plant instrumentation, have been found to be acceptable without modification. Plant emergency operating procedures have been updated to account for changes in the accuracy of affected instrumentation.

Steam Line Break Outside Containment

The mass and energy releases for main steam line breaks (MSLBs) outside containment have changed as a result of the Cycle 2 core design and other changes (see DCN W-39293-A) which has resulted in revision of the temperature environment in the main steam valve vaults (MSVVs) following a MSLB. The change in environmental temperatures in the MSVVs results in revisions to thermal lag calculations and material aging calculations for safety-related electrical equipment in the MSVVs. Junction/splice boxes have been thermally insulated, as necessary to ensure electrical cable temperature remains below their qualification temperature during the MSLB (see DCN W-39538-A, which is a contingency DCN and is to be worked only if junction/splice boxes below a minimum size are found in the MSVVs). All safety-related equipment has been determined to remain environmentally qualified.

Hydrogen Generation

The 1000 EFPD source term has been used in the calculation of the hydrogen generation due radiolysis. In addition, the revised hydrogen generation calculation reflects the hydrogen generation due to the mass of aluminum, zinc and zirconium in the Cycle 2 core. The results of the calculation demonstrate that the post-LOCA hydrogen generation for the Cycle 2 core is still less than the

analyses and evaluations demonstrate that (with minor plant modification, see A.1) the affected structures, systems and components in the MSVVs continue to be able to perform their intended functions and therefore, this change does not constitute a USQ.

Hydrogen Generation

The increase in post-accident hydrogen generation, as a result of the Cycle 2 core design, has been factored into design basis analyses and evaluations. The results of these revised, design basis analyses and evaluations demonstrate that the post-LOCA hydrogen generation for the Cycle 2 core is still less than the current design basis hydrogen generation and therefore, this change does not constitute a USQ.

Tritium Production

The changes in tritium production and transport to the reactor coolant, as a result of the Cycle 2 core design, has been updated. The updated estimates of tritium production and transport do not affect design basis analyses and evaluations and therefore, this change does not constitute a USQ.

current design basis hydrogen generation.

Tritium Production

The increases in cycle length, reactor coolant boron concentration during operation and proposed fuel clad material change to ZIRLOTM affects tritium production and transport to the reactor coolant. Tritium data for Cycle 2 and future cycles has been updated.

SA-SE Number *WBPLMN-97-032-0*

Affected Documents	*Document Type*	*Safety Assessment Title*	*Implementation Date:*
DCN S-39271-A FSAR Change Pkg. 1471	DCN	Various sub-sections of section 10.4 of the FSAR and various system descriptions and/or design criteria are revised resulting from an open review as required by WBPER970051.	9/9/97

Description of Change, Test, or Experiments

This DCN and associated FSAR Change Package #1471 revises various sections of FSAR section 10.4 and various system descriptions and/or design criteria to provide editorial changes, clarifications, and corrections resulting from an open review as required by WBPER970051. For detailed changes to the FSAR see the SA tables which list reference to the item number, associated FSAR section and page; see also the specific change in the FSAR Change Package #1471. This DCN is a documentation change only, there are no plant modifications required by this DCN. Therefore, an SE is required by the SA review. However, since all 68 FSAR changes are clarifications and do not involve plant modifications only items 36, 40, and 55 (see SA Tables - pages 4, 5, and 6) will be addressed in the SA Checklist and the SE. Items 36, 40, and 55 are being addressed because a loss of one of these pumps can cause a plant shutdown or reduction in power output. A description of these items follows:

Item 36 - "However, with all feedwater heaters in service and all heater drains being pumped forward, 100% load can be maintained with only two condensate booster pumps running. (this description is deleted)

Item 40 - With the No. 7 heater drains cascading to the main condenser and all heater banks in service, loss of any one hotwell pump and/or one condensate booster pump will not affect the reactor coolant system, because the unit will still be capable of maintaining 100% guaranteed load." (description deleted)

Item 55 - "The main feed pump turbine condenser drains are equipped with two 100% capacity pumps which take suction from a single drain tank." (100% capacity is deleted)

Safety Evaluation Summary

DCN S-39271-A and associated FSAR Change Package #1471 does not affect any FSAR Chapter 15 fault or operational transient evaluations. Nor does it increase the probability of occurrence or consequences of the analyzed event of a Loss of Normal Feedwater. Neither the probability of occurrence or consequences of a radwaste processing system leakage event is increased. The possibility for an accident or malfunction of a different type than any previously evaluated in the FSAR is bounded by the plant procedures for restoration of main feedwater (worst case scenario) or operator action to reduce power level to restore steam generator water levels in the event of a lost hotwell and condensate booster pumps. No Tech Spec margins of safety are reduced. This change does not adversely affect any equipment important to safes, either directly or indirectly. Therefore, on the basis of the evaluation of effects, it is concluded that DCN S-39271-A is acceptable from a nuclear safety standpoint and no unreviewed safety question exists.

The Watts Bar Safety Evaluation Report (SER) and Supplements 1-20 (specifically sections 10.2 through 10.4.9) have been evaluated for impacts. DCN S-39271-A does not impact the NRC's understanding of the design and operation of Watts Bar Nuclear Plant as described in the FSAR.

SA-SE Number *WBPLMN-97-033-0*

Affected Documents	*Document Type*	*Safety Assessment Title*	*Implementation Date:*
DCN W-39288-A	DCN	Replace Check Valve with Stainless Steel Valve With No Piston Spring and Control Pressure Increased to 4 psi	10/12/97

Description of Change, Test, or Experiments

Nitrogen (N2) is supplied to the Reactor Coolant Drain Tank (RCDT) either as cover gas or to purge the tank. During purging of the RCDT, check valve 1-CKV-077-0505 was determined to restrict the N2 flow. This valve is made for either a liquid or gas application but for operating pressure greater than the existing pressure of 1.26 psi. The RCDT capacity is 350 gallons or 46.79 ft3 and the N2 flowrate is as follows:

Pump	Operating Capacity (gpm)	Time to DrainTank (min)	N2 Required (cfm)
A	50	7	6.68
B	150	2.33	20.05
A & B	200	1.75	26.74

A spare check valve was informally tested and with the piston spring installed the cracking differential pressure is 2 psi. The valve did not achieve the full open position even at a differential pressure of 6 psi. The new valve, the cracking pressure should be approximately 0.5 psi and greater than approximately 1 psi to be fully open. Since new valve should achieves the fully open position at approximately 1 psi differential or greater, the Cv of 6 specified for the existing valve on the vendor's drawing can be taken when N2 is supplied at 4 psi. A Cv of 6 equates to approximately 30 ft equivalent length of pipe. Therefore, the total equivalent length between 1-PCV-077-0158 and the RCDT (including 1-CKV-077-0505) can be conservatively taken to be 300 ft. The N2 flowrate has been computed for a differential pressure of 4 psi and the N2 flowrate available exceeds the flowrate required to fill the space evacuated due to the maximum RCDT pump(s) flowrate out of the tank

Design Change Notice (DCN) NO. 39288-A is replacing check valve (1-CKV-77-0505) with a similar type valve, changing the body material from carbon steel (CS) to stainless steel (SS), the new valve should not have a piston spring, and increasing the N2 control pressure control to 4 psi for the 1-PCV-077-0158. These changes will insure that sufficient N2 volume is supplied to the RCDT.

Safety Evaluation Summary

The RCDT, associated components, piping, and valves are located in the Reactor Building on elevation 702 at azimuth 270°. This equipment does not perform a primary safety function (except for containment isolation valves which have not been affected), are installed in a Seismic Category I structure, and are not used during any accident. The Chapter 15 accident analysis identifies an unexpected and uncontrolled release of radioactive xenon and krypton fission product gases stored in a Waste Gas Decay Tank as a consequence of a failure of a single Waste Gas Decay Tank or associated piping. This modification will not increase the consequences of an accident previously evaluated, and is bounded by the existing analysis of the Waste Gas Decay Tank. This DCN does not change the logic or function of any system that is important to safety. A review of the detailed changes leads to the SE conclusions that this change is safe and does not constitute an USQ.

FSAR Section 11.2.4 identifies equipment faults which could occur with moderate frequency, including fuel cladding defects in combination with malfunctions in the Liquid Radwaste Processing System such as pump or valve failures or evaporator failures. Chapter 15 accident analysis identifies an unexpected and uncontrolled release of radioactive xenon and krypton fission product gases stored in a Waste Gas Decay Tank as a consequence of a failure of a single Waste Gas Decay Tank or associated piping. This DCN is not associated with the equipment that could cause these events. Also, this change is not associated with the protective features used to detect and mitigate the effects of these events. The equipment involved in the modification do not interface with any equipment whose malfunction could result in an accident which has been evaluated in the FSAR. This DCN does not change or affect the design basis for any system that is important to safety.

These changes do not affect any equipment required for safe operation or shutdown. In the event of a DBA, all safety related equipment is expected to operate as designed to limit the consequences of the DBA. No new potential single failures of existing components will occur as a result of replacing the CKV and increasing the N: supply pressure to the RCDT. This DCN does not add any additional or different types of failure modes that have not been addressed in the FSAR. Neither will this change cause this system or any system important to safety to fail to fulfill its functional requirements. The RCDT, its associated components, and piping do not perform any accident mitigation function except for containment isolation valves which have not been affected. The accidents and/or malfunctions associated with the Radwaste system is a failure of Waste Gas Decay Tank or associated piping and failure of Radwaste components. Although this change does affect Radwaste components, this equipment is not used in the mitigation of these accident/malfunctions and does not change the radiological consequences of an accident previously evaluated in the FSAR. A malfunction of the CKV to not open would not prevent operations from purging the RCDT since, as an alternate, the Waste Gas vent header could be isolated with valve 1-ISV-077-0593 and N2 supplied to the RCDT through test connection 1-ISV-077-0592. If the CKV fails to close, back flow would still be prevented by either valve 1-FCV-77-20 or 1-PCV-77-158 which are upstream of the CKV.

These changes do not reduce the margin of safety identified in the applicable Technical Specifications. These changes do not prevent any component from performing its function as described in the Technical Specifications. The ODCM limits for releases from either the Liquid Radwaste Processing System or the Gaseous Waste System are not revised or challenged by these changes.

SA-SE Number *WBPLMN-97-039-0*

Affected Documents	*Document Type*	*Safety Assessment Title*	*Implementation Date:*
DCN M-39179-A FSAR Change Pkg. 1480	DCN	New Relief Valve Within the Isolated Boundary of Containment Penetration	9/24/97

Description of Change, Test, or Experiments

The Safety Injection (SI) System accumulator fill and check valve leak test line penetrates primary containment through penetration no. X-30. Currently System Description Document (SDD) N3-63-4001, section 4.8 requires that the penetration be drained after each use to prevent temperature-induced overpressurization in the event of a Loss of Coolant Accident (LOCA) or Main Steam Line Break (MSLB). The piping inside containment would be subjected to an environmental temperature of approximately 235 °F following a LOCA or MSLB accident. The current SDD requirement is implemented by system operating instructions. DCN M-39179 adds new relief valve 1-RFV-63-0028 within the isolated boundary of containment penetration X-30, thereby eliminating the need to drain the subject piping after each use. The new relief valve will now provide the required thermal relief protection automatically in the event of a LOCA or MSLB. This SA/SE addresses the scope of system operating instruction changes necessary to remove steps that currently require draining of penetration X-30 after each use.

This SA/SE also addresses the following FSAR changes which are issued in support of DCN 39179-A. Table 3.9-25 is revised to identify 1-RFV-63-28 as an active valve. Table 3.9-26 is revised to identify that 1-RFV-63-28 is Category A & C for inservice inspection. Table 6.2.4-1 is revised to reflect that a 10 CFR 50 App J Type A and Type C test will be required for 1-RFV-63-28. FSAR page 6.2.4-2 is revised to further clarify that relief valves can be used as containment isolation valves provided that the setpoint is greater than 1.5 times the containment design pressure. Table 6.2.6-2a is revised to include 63-28 as a containment isolation valve requiring Type C testing for penetration X-30. Table 6.2.6-3 is revised to reflect that penetration X-30 will now have a status of "Normal Lineup" with the implementation of DCN 39179-A. Table 3.9-20 is revised to identify the new relief valve for penetration X-30.

Safety Evaluation Summary

The changes of DCN 39179-A and FSAR Change Package 1480 do not affect any FSAR Chapter 15 fault or operational transient evaluations. The changes to add new relief valve 1-RFV43-28 comply with all design and licensing basis requirements for the Containment Isolation System. The relief valve has no design function for normal plant operation. The relief valve's design function is to prevent over-pressurization in the event of a LOCA or MSLB and; therefore, does not have the potential to increase the probability of an accident previously evaluated in the FSAR. The new relief valve and its associated inlet and discharge piping does not increase the probability of occurrence of a malfunction of the containment isolation function performed for penetrations X-24 and X-30. Nor is the probability of a malfunction of any important to safety equipment increased during implementation of the modification. Since containment integrity and containment isolation capability is unaffected by the addition of relief valve 1-RFV43-28, the radiological consequences of an accident previously evaluated in the FSAR is not increased. The modification is in compliance with the FSAR described provisions for assuring redundant containment isolation such that the radiological consequences of malfunctions of equipment important to safety previously evaluated in the FSAR are not increased. Since the new relief does not perform any function during normal plant operation, the potential for perturbations to normal plant operation that could lead to an accident of a different type will not exist. The credible malfunction mechanism for 1-RFV43-28 is a failure in the open position. This malfunction of the valve does not compromise the present redundancy in design that exists for penetrations X-30 and X-24 such that addition of 1-RFV-63-28 does not create a possibility for a malfunction of a different type than any evaluated previously in the FSAR. The safety margins for the Emergency Core Cooling System as defined in Tech Spec 3.5 are not reduced. In addition the Bases for Tech Spec 3.6.3, Containment Isolation Valves, have been reviewed to assure that no safety margins are affected. Therefore, on the basis of the evaluation of effects, it is concluded that DCN 39179-A and associated FSAR Change Package 1480 are acceptable from a nuclear safety standpoint and no unreviewed safety question exists.

SA-SE Number *WBPLMN-97-041-2*

Affected Documents	*Document Type*	*Safety Assessment Title*	*Implementation Date:*
DCN M-39191-A	DCN	Replacement of Main Feedwater Pumps and Standby Main Feedwater Pump Recirculation Valves with the Control Components DRAG Valves	10/13/97

Description of Change, Test, or Experiments

Currently the WBN MFP and SBMFP recirculation (mini flow) valves, 1-FCV-3-70 & -84 & - 208, are Valtek model 9376-1 and -2 valves. Operations has had difficulty in operating these valves in the automatic mode due to valve throttling concerns and the inability for tight shutoff at system operating pressure (due primarily to a flow under the seat valve design). Therefore, Operations has chosen to abandon the automatic mode and operate the valves in a manual mode which is considered a Work Around". This DCN will restore the valves' capability to operate in the automatic mode by replacing the valves with the CCI DRAG valves and changing the associated control loops. The purpose of the recite valves is to provide a minimum required flow of 4000 gpm (MFPs) and 1500 gpm (SBMFP) during low flow conditions (startup and feedwater isolation) and discharge back to the Main Condenser. As such, they can see a pressure drop from approximately 1200 psig to 1300 psig (feed pump discharge pressure) to the condenser vacuum of 1 psia. The Valtek valves have proven to be inadequate for this severe service. The replacement CCI DRAG valves are designed specifically for this severe service application of high pressure drop and hot water flashing within the valve body to a water/steam mixture. The CCI valves are also designed for opening against a high differential pressure. Also there is an instrumentation and control (I&C) concern associated with putting these valves back in the automatic mode. This concern is the ability to open the recirc valves for a plant transient involving main feedwater isolation. The necessary valve action must be rapid to help relieve any downstream pressure spikes caused by this isolation and prevent damage to the #1 feedwater heater channel relief valves. This can best be accomplished by the addition of quick-exhausting volume boosters on the new positioners being purchased as part of the valve replacement. The feedwater isolation signal will be tied to the flow indicating controller from feedwater pump trip circuits. A feedwater isolation signal will open the process input circuit to the recirc valve controller causing the output signal to the flow modulator to instantaneously go to the full open position. The valve will remain in the full open position as long as the feedwater isolation signal is present and will revert back to modulated control as soon as the feedwater isolation signal is reset. The controller can still be placed in manual operation to manually control the recirc valve if so desired. The feedwater isolation signal will be derived from spare contacts on the feedwater isolation separation relays, (HWPA & HWPB), and wired to a spare relay in the auxiliary relay panel 1-R-75. A normally closed contact from this spare relay will be cabled to each controller process input circuit in the BOP instrument racks 1-R-128, 1-R-131, & 1-R-141. Therefore, a new signal cable will have to be routed from 1-R-75 to each of the BOP racks. In addition, there will be terminal strip re-wiring required in each BOP rack to wire in the normally closed relay contact into the signal path between the square root flow modifier and the flow indicating controller for each valve control circuit. Also, a normally closed contact from the 1-R-75 spare relay will be wired to the control circuit for FSV-3-208 to keep the solenoid from energizing upon feedwater isolation. This will necessitate that a control signal cable be routed between panel 1-R-75 in the Auxiliary Instrument Room to JB 652 in the Turbine Building, elevation 729. This will allow the valves to open rapidly for a main feedwater isolation condition, but for all other plant transients the valves will gradually move toward the open or closed position for lower or higher flow conditions.

The Watts Bar Safety Evaluation Report (SER) and Supplements 1-20 have been evaluated for

Safety Evaluation Summary

Implementation of this DCN does not affect any FSAR Chapter 15 fault or operational transient evaluations. Nor does it increase the probability of occurrence or consequences of the analyzed event of a "Major Rupture of a Main Steam Line" which assumes complete Feedwater isolation. The existing Chapter 15 safety analyses take credit for the closure of the MFIVs and MFRVs and associated bypass valves as well as shutdown of all Main Feedwater Pumps, Condensate Booster Pumps, and Con Demin Pumps for any event which generates a feedwater isolation signal. Neither the probability of occurrence or consequences of a radwaste processing system leakage event is increased. The possibility for an accident or malfunction of a different type than any previously evaluated in the FSAR remains unchanged. This DCN does not alter the existing safety analyses. Opening the Main Feed pumps recirc valves does not change any feature or function of the Feedwater Isolation logic. Even though the Feedwater Isolation signal is used to open the recirc valves, it is not a part of the safety grade Feedwater Isolation process. The addition of the Feedwater Isolation feature to the Main Feed pump recirc valves is purely an enhancement to prevent overpressurization of the main feed pump discharge piping during pump coastdown for this event. The addition of the feedwater isolation feature to the main feed pump recirc valves does not adversely impact the safety function performed by the feedwater isolation signal. No Tech Spec margins of safety are reduced. This change does not adversely affect any equipment important to safety, either directly or indirectly. Therefore, on the basis of the evaluation of effects, it is concluded that implementation of this DCN is acceptable from a nuclear safety standpoint and no unreviewed safety question exists.

impacts. The plant modifications created by this DCN do not impact the NRC's understanding of the design and operation of Watts Bar Nuclear Plant as described in the FSAR because the function of the recirc valves is to recirc a deadheaded pump and this function is not altered by this DCN. Specifically, the SER (section 10.4.7) does not address the operation of the Feed Pump recirc valves.

SA-SE Number *WBPLMN-97-044-0*

Affected Documents	*Document Type*	*Safety Assessment Title*	*Implementation Date:*
DCN S-39417-A FSAR Change Pkg. 1476 PAI 4.01, R6	DCN	DCN revises the Radiation Monitoring System Design Criteria, the Main Steam System Description, the Condensate System Description, and the FSAR to credit use of the Steam Generator blowdown monitors, Main Steam monitors, a portable monitor and sampling	12/30/97

Description of Change, Test, or Experiments

Prior to achieving a vacuum in the Main Condenser(6.5 inches Hg absolute), the high condenser vacuum exhaust (CVE) flow drives water into the CVE radiation Monitors. Since the CVE Radiation Monitors contain components sensitive to water(i.e., carbon pump vanes and charcoal filters), the monitors cannot be operated until a vacuum is achieved in the condenser. DCN S-39417-A is issued to credit use of the Steam Generator Blowdown Radiation Monitors, a portable monitor and sampling in accordance with the Offsite Dose Calculation Manual (ODCM) in lieu of the using the CVE Radiation Monitors to detect a Steam Generator Tube Leak and to assess radioactive effluents through the condenser vacuum exhaust prior to achieving a vacuum in the Main Condenser. In addition, this change credits use of the Main Steam Radiation Monitors and sampling as required by Emergency Plan Implementing Procedure (EPIP) 16 to quantify radioactive effluent release from the condenser vacuum exhaust in the event of a steam generator tube rupture during the time the CVE Monitors are not operating. Revision of the ODCM(PAI 4.01) to indicate that the CVE Monitors are not required to operate prior to achieving a vacuum in the main condenser is addressed by this SA/SE. Revision of other plant procedures resulting from this change will be addressed by separate SA/SE.

The purpose of the CVE monitors are as follows:

Sampler 1-RE-90-129(particulate & iodine) and Monitor 1-RE-90-II9(Gas) -- Monitors effluents during normal operations.

Monitor 1-RE-90-119 -- Provide indication of a steam generator tube leak.

Monitor 1-RE-90-404 -- Monitors plant effluents subsequent to a steam generator tube rupture.

DCN S-39417-A revises the Radiation Monitoring System Design Criteria (WB-DC-40-24), the Main Steam System Description(N3-1-4002), the Condensate System Description (N3-2-4002), and FSAR sections 11.4.2.2.2 and 5.2.7.4.2 to permit use of the Steam Generator blowdown monitor 1-RE-90-120, 1-RE-90-121 or portable monitor, to detect a steam generator tube leak, and use of the Main Steam Radiation Monitors 1-RE-90-421, 422, 423, & 424, and sampling per the ODCM and procedure EPIP-16 to assess radioactive effluent releases through the CVE. This process is applicable only when the CVE Radiation Monitors are not operating during the time a vacuum is being established in the Main Condenser(until the condenser pressure is 6.5 inches Hg absolute). As a result of this change, ODCM table 1.1-2 is revised to indicate the CVE Radiation Monitors are not required to function prior to achieving a vacuum in the main condenser.

If a Steam Generator blowdown monitor (or portable monitor if the Steam Generator blowdown monitors are not available)indicates a steam generator tube leak, the radioactive particulate, iodine and gas effluent through the CVE will be assessed under the provisions of tables 2.2-1 and 2.2-2 in

Safety Evaluation Summary

SER 10.4.2 and 11.5.1 state that the CVE Radiation Monitors continuously monitor the effluents from the CVE. Licensing condition 6(a) and 6(b)/NUREG 0737 required continuous monitoring of plant effluents. Regulatory Guide 1.97 requires continuous monitoring of noble gas activity from the CVE. To meet the requirements of Regulatory Guide 1.21 and NUREG 0800, continuous monitoring is required for major or potentially significant release paths which are essentially uninterrupted for extended periods of time (i.e., continuous).

This change permits use of alternate equipment (Steam Generator Blowdown or portable monitor, and Main Steam Radiation Monitors) and sampling procedures (ODCM and EPIP-16)for detection of a steam generator tube leak and assessment of potential radioactive effluent releases through the CVE in lieu of use of the CVE Radiation Monitors prior to establishing a vacuum in the main condenser. The Steam Generator Blowdown Radiation Monitors are continuous monitors used for steam generator tube leak detection and are currently discussed in FSAR Section 5.2.7.4.4; as indicated in this FSAR section, the Steam Generator Blowdown Monitors are continuous monitors and can detect a steam generator tube leak within the regulatory guidelines as established in Regulatory Guide 1.45 (i.e.,1 GPM in one hour); if these monitors are unavailable, a portable monitor will be used to identify a steam generator tube leak. This portable monitor will be used each hour to monitor the CVE to ensure a steam generator tube leak is detected within one hour. The procedures for assessment of radioactive effluents during normal operations currently exist in tables 2.2-1 and 2.2-2 of the ODCM. The Main Steam radiation Monitors, which are discussed in FSAR Section 11.4.2.2.7, are continuous monitors designed for identification and monitoring of a steam generator tube rupture accident. Determination of plant effluents by sampling during accident conditions is currently in EPIP-16.

Use of alternate continuous radiation monitors and/or a portable monitor in conjunction with sampling is an acceptable alternative to continuously monitoring with the CVE Radiation Monitors. Note the ODCM currently permits sampling of the CVE if the CVE monitors are inoperable.

This change does not add new permanent plant equipment nor make physical changes to existing equipment. Since alternate continuous radiation monitors and/or a portable monitor in conjunction with sampling are used to perform the functions of the CVE monitors, this change does not increase the probability of occurrence or consequences of accidents or malfunctions currently evaluated in the FSAR, does not create a possibility of a different type of accident or malfunction evaluated in the FSAR, and does not reduce the margin of safety identified in the applicable Technical Specifications. Consequently, DCN S-39417-A does not result in an Unreviewed Safety Question.

the ODCM. If a portable monitor is used in lieu of a Steam Generator blowdown monitor, the detector will be placed in close proximity to the CVE flow on the roof of the turbine building each hour to insure detection of a steam generator leak within one hour. The alarm set point for the portable monitor will be consistent with the set point of the CVE monitor 1-RE-90-119, which is currently set at two times expected background. The sensitivity of the portable monitor will be such that it can detect activities resulting from a 1 GPM primary to secondary leak within one hour. Consequently, the portable monitor is adequate to detect a steam generator tube leak of one GPM within one hour. In the unlikely event of a steam generator tube rupture during this time, the tube rupture will be identified by the Main Steam Radiation Monitors 1-RE-90-421,422,423 & 424, and radioactive effluents through the CVE will be determined by sampling per the requirements of EPIP-16. In addition, the steam activity, as determined by the Main Steam Radiation Monitors can be used in conjunction with the quantity of steam entering the condenser to determine the amount of radioactivity released through the condenser vacuum exhaust.

Particulate/iodine Sampler 1-RE-90-129 is not required to function subsequent to an accident per R.G. 1.97 Table 3 notes on type E variables.

This SE also addresses changes being made to the ODCM to change the operability requirements for the 1-RE-90-119 monitor and 1-RE-90-129 sampler. The change will revise the wording for applicability note (3) on Table 1.1-2 to state that these pieces of equipment are not required until a vacuum is fully established in the Main Condenser. This revision will not change the sampling requirements for the CVE used to quantify radioactive releases to unrestricted areas, therefore the accuracy and reliability of effluent, dose and setpoint calculations described in the ODCM is not impacted. The level of effluent control required by 10CFR20.1302, 40CFR190, and Appendix I to 10CFR50 is not reduced by this change since the functions performed by the 1-RE-90-119 monitor during this mode of operation will be provided by alternate means.

SA-SE Number *WBPLMN-97-050-0*

Affected Documents	*Document Type*	*Safety Assessment Title*	*Implementation Date:*
DCN S-39440-A FSAR Change Pkg. 1487	DCN	System 30 System Description Document revised to clarify the ABGTS Differential Pressure	12/5/97

Description of Change, Test, or Experiments

The EGTS air cleanup subsystem is a redundant, shared airflow network having the capability to perform two functions for the Containment Annulus during a LOCA. One function is to keep the Annulus air pressure lower than atmospheric and that of any adjacent space. The second function is to remove airborne particulates and vapors that may contain radioactive nuclides from air drawn from the Annulus. The ABGTS is a fully redundant air cleanup network provided to reduce radioactive nuclide releases from the ABSCE during accidents. To accomplish this function, it draws air from all parts of the AB and the CDWE Building to establish a negative pressure region in which virtually no unprocessed air passes to the atmosphere. The EGTS and the ABGTS are controlled by their respective DP controllers. Setpoints for the Annulus and the ABSCE controllers are -1.45 inches wg, and -0.37 inches wg, respectively. The ABSCE DP (nominal -0.25" wg) must not become more negative than that of the Annulus (nominal -0.5" wg) to preclude the possibility of drawing potentially more contaminated air from the Annulus into the AB.

Both Trains of the EGTS and the ABGTS start automatically on an accident signal. WBPER970206 identified a condition, during a pressure test when both ABGTS fans were operated concurrently with a vacuum relief damper failed closed (to simulate a single worst-case failure) and the ABSCE DP was measured at -1.09" wg. This condition, the ABSCE DP more negative than the ABGTS lower analytical limit of -0.5" wg, in the stated failure mode, is evaluated by calculations EPM-MMA-121889 R5 and EPM-WJK-041592 R9, and found to be acceptable. These Calculations show, by analysis, that the Annulus DP will always remain more negative than -1.09" wg at the same elevation as that of the AB where the ABSCE DP is measured/controlled. The calculations conclude that the Annulus DP will remain more negative than that of the ABSCE. Calculations TI-630 R7 and EPM-WVC-101089 R23 are issued to document the failure mode, and to document the DP of -1.09" wg and provide input to I&C calculations, respectively. The calculation 0-PDT-65-80 R6 is issued to change the EGTS DP controller (upper) allowable value to -1.17" wg. SDD N3-30AB-4001 R4 is revised to clarify that it is acceptable in certain operational conditions for the ABSCE DP to go more negative than -0.5" wg. Design Criteria WB-DC-40-42 R2 is revised to express the need to use more conservative outside maximum/minimum temperature values than those listed on the Environmental Data drawings for special analyses. The FSAR is revised to insert the revised FMEA calculation pages in Table 6.2.3-3, correct the duration of "abnormal condition" in Sections 3.11.2.1 and 9.4.1.3, clarify text and eliminate redundancies in Sections 6.2.3.2.1, 9.4.2.1 and 9.4.3.1, and insert the phrase "except when under administrative control" to Section 3.8.4.1.1 to permit opening both doors of an ABSCE airlock during normal operation to perform maintenance and other work on the boundary doors.

Safety Evaluation Summary

This change further evaluates the failure mode for the ABGTS, consisting of the single failure of a vacuum relief damper while running both ABGTS fans during the initial 30 minutes of operation following an accident. This could result in the ABSCE DP going more negative than -0.5" wg. The acceptability of the ABGTS to drawdown the ABSCE to more negative pressures than -0.5" wg in certain operational conditions without adversely affecting the functional integrity of the EGTS is documented in calculations EPM-MMA-121889 R5 and EPM-WJK-041592 R9. These calculations show that the Annulus DP is maintained more negative than the ABSCE DP during all operational conditions, including the worst-case single failure mode described above. Therefore, any leakage between the ABSCE and the Annulus following a LOCA will be into the Annulus. This change also revises the FSAR to insert the revised FMEA calculation pages in the ABGTS FMEA tables, correct the duration in the definition of "abnormal environmental condition", clarify text for the ABSCE and the AB HVAC and eliminate redundancies, and insert the phrase "except when under administrative control" to ABSCE Boundary Doors section to permit opening of both doors of an ABSCE airlock during normal operation to perform maintenance and other work on the boundary doors. Revising the FSAR to reflect the analysis results which confirm the operability of the ABGTS and the EGTS within their design limits considering all failure modes, and administrative FSAR changes do not increase the probability of occurrence or consequences of accidents or malfunctions currently evaluated in the FSAR, does not create a possibility of a different type of accident or malfunction evaluated in the FSAR, and does not reduce the margin of safety identified in the applicable Technical Specifications. Consequently, DCN S-39440-A does not result in an Unreviewed Safety Question.

SA-SE Number *WBPLMN-97-052-2*

Affected Documents	*Document Type*	*Safety Assessment Title*	*Implementation Date:*
DCN W-39382-A	DCN	Extend the Safety Injection System high point vent piping to the floor	10/6/97

Description of Change, Test, or Experiments

This modification reconfigures the SIS vents in the fan and accumulators rooms and the vent in the valve vault in close proximity to the BIT. The original configuration has a short section of pipe, a normally closed valve and a blind flange. This arrangement provides for double isolation but is not located physically for operational convenience. The modified configuration has the vent piping run to the floor level and terminates in two normally closed valves, at which point a "B" to "G" class break is taken, and ends in a short section of pipe with a threaded pipe cap. This provides for operational convenience and reduced ALARA concerns.

During the performance of Surveillance Instruction 1-SI-63-IOA, a monthly Tech Spec requirement, a considerable amount of effort, time and manrems expenditures savings result, from this modification, during the performance of the SI. This modification eliminates the need for ladders or scaffolding and places the valves in an area that is much less of an ALARA concern.

The modified configuration retains an existing vent valve at the upper elevation and adds two new valves near the floor elevation. The addition of the two valves does not change the size or function of the vent. The additional amount of water that will be required to be drained to ensure that the SIS piping is water solid will be two or three gallons based on the ¾ inch pipe size and length of run added. This is not a undue burden on the drainage system or the radwaste system.

As the piping is now identified as Class B, the additional length of pipe and added valve are Class B. The additional valve allows a class break at the end of the second isolation valves. This eliminates the need for a blind flange to provide the second isolation before a class break can be taken. A short pipe nipple with a threaded cap is provided for convenience of attaching the drain hose during the SI performance.

Safety Evaluation Summary

The usage of valves to achieve double isolation and thereby, a class break from nuclear safety to non-nuclear safety is allowed per ANS-18.2 (now ANS 51.1). This allowance provides assurance that the usage has been evaluated and found acceptable. The extended piping is fully qualified to the requirements of ASME Section III, and provides no additional safety hazards from the original configuration.

SA-SE Number *WBPLMN-97-053*

Affected Documents	*Document Type*	*Safety Assessment Title*	*Implementation Date:*
DCN M-39380-A FSAR Change Pkg. 1484	DCN	Enhancement of the Drainage System from the Main Feedwater Pump Turbine Drain Tank to the Condenser	10/12/97

Description of Change, Test, or Experiments

The main feed pump turbine condenser drain tank receives the drains from both main feed pump turbine condensers. Normal tank level control is maintained by two pumps which take suction from the tank and discharge through a level control valve (LCV-6-206) into the main condensed. A bypass system is also provided in the current design which is a 4-inch line with a level control valve and also discharges into the main condensed. The tank is equipped with a single hi level switch which opens the bypass valve and alerts the operators.

This modification provides for a new gravity drain system from the drain tank to the main condenser. A control valve in conjunction with two isolation valves is routed to the hotwell section of the main condenser. This drain system will function as the normal tank level control and replace the existing pump drainage system. The pumps have experienced difficulties handling the mass flow under maximum operating conditions. The new gravity drain is sized to handle the mass flow, full capacity operating conditions. The reliability of the gravity drain (passive system) as opposed to the pumped drainage (active system) is increased due to the decreased complexity of the passive system.

In the unlikely event where failure of the new level control valve in the 8-inch drain line occurs, the valve will fail in the open position and will maintain the flow path to the Main Condenser, until a flow path can be established through the drain pump loop.

The single Hi-level switch which opens the existing bypass valve and annunciates Hi-level will also be modified by this DCN. The hi-level will continue to be annunciated, however a separate Hi-Hi-level switch will be added to operate the bypass valve. Failure of the original single switch design would have prevented both the opening of the bypass valve and Hi-level annunciation.

Although the change has no nuclear safety significance an FSAR change is necessary because the original drain system is detailed in the FSAR text. FSAR Change Package 1484 describes the modification.

Safety Evaluation Summary

The function of tie main feed pump turbine condenser drain tank level Control system is not modified by this DCN, but is enhanced to be more reliable.

The safety Classification of this system is non nuclear safety related and is designed in accordance with ANSI B31.3 Code requirements. No Tech Specs requirements are involved and no accident parameters are affected; therefore, nuclear safety is not affected.

The piping system modified by the subject DCN is not important to safety and is not safety related but is described in the FSAR. A non-significant change to the FSAR was necessary (see FSAR CP 1484). Based on the non-safety related function of the piping, the conclusion that nuclear safety is not reduced is justified. Furthermore, the changes implemented by this DCN do not increase the probability of any secondary side events, such as main feed pump trip or turbine trip.

SA-SE Number *WBPLMN-97-060-1*

Affected Documents	*Document Type*	*Safety Assessment Title*	*Implementation Date:*
DCN 39375-B	DCN	6" Gland Seal Steam Condenser Spillover Pressure Control Valve and Bypass Valve	10/7/97

Description of Change, Test, or Experiments

DCN 39375-B implements the partial corrective action of WBPER961031. WBPER961031 was initiated due to the generic review of SQ961725 PER. Based on engineering analysis, this DCN implements design changes necessary to the main turbine gland steam spillover system piping and valves 1-PCV-47-190, 1-PCV-47-191, 1-PCV-47-193, & 1-ISV-47-708) to provide automatic control of HP Turbine gland pressure under all load conditions. The solution to the problem of inadequate capacity is to replace the valves and piping in the control valve station and the discharge piping from the control valve station with 8" valves and piping; change 1-PCV-47-190, the isolation valve upstream of 1-PCV-47-193, from a globe valve to a gate valve, renamed 1-FCV-47-190; and reroute the control station discharge piping from the No. 7 heaters to the main condenser, providing lower line losses and lower downstream pressure in which to discharge. The larger control valves and piping, decreased line losses, and discharge to a lower pressure will provide adequate capacity for the gland steam spillover system to operate under all conditions.

Additionally, the controls are being modified for valves 1-PCV-47-189 and -193 such that the pressure loop 1-PT-47-189 will now be used to control the action of both valves. This will eliminate tolerance differences and variations in pressure at the sensing location and result in smoother switchover from 1-PCV-47-189 to 1-PCV-47-193 at around 20 psia.

Since the piping being replaced falls under the scope of the Flow Accelerated Corrosion (FAC) Program, the piping will be replaced using 2 % Chrome 1% Molybdenum pipe/fitting material. This material is more resistant to erosion than the carbon steel material originally installed. The piping being replaced falls in the priority 2 area of the FAC Program, but will be replaced under this DCN due to cost and convenience considerations. Since the discharge piping is being rerouted, a revised piping analysis will be performed and a new support configuration will be required.

The existing discharge piping to the No. 7 heaters will be capped near the control valve station underneath the 729.0 floor where it is being replaced and rerouted, and the abandoned piping will be left in place. Removal would be unnecessary and cannot be economically justified. The end of the abandoned spillover discharge piping where it intersects with the piping going to heater 1A-7 will have a section cut out, and both open ends will be capped.

Replacement of 1-PCV-47-190, 1-PCV-47-191, 1-PCV-47-193, 1-ISV-47-708, and the associated piping may be made in Mode 5 or 6. Alternatively, the replacement may be made in Mode 3 or 4 with vacuum broken on the Main Condenser and "steam blocking clearance" in place (Main Steam Isolation Valves closed and Auxiliary Boiler steam supply to seals isolated).

Revision 1 of this SA/SE is to address the DCN being staged into two stages. All work except installing, calibrating and testing of the positioner for 1-PCV-47-189 will be in stage 1. Installing, calibrating and testing of the positioner for 1-PCV-47-189 will be in stage 2.

Safety Evaluation Summary

The items being replaced are part of the Turbine Gland Sealing System. This system prevents leakage of air into the turbine casing and prevents the leakage of steam into the turbine room when the turbine casing is pressurized. The system is designed in accordance with ANSI B. 1.1, is not safety related, and is not required to operate to safely shut down the plant following any initiating event. All the affected equipment is located in the Turbine Building. During normal operation, the Turbine Gland Sealing System is a source of radioactive material not normally considered part of the radioactive waste system. The modifications implemented by DCN 39375-B are improvements to the existing system and do not change the operation of the system or add equipment of a different type which could contribute to an accident, nor do they affect the plant leakage from the Turbine Building Ventilation System and Turbine Gland Sealing System as identified in Chapter 11 of the FSAR.

State of the art instrumentation will result in higher reliability and less likelihood of turbine trip and reactor trip above 50% power. The single sense point (the same transmitter is used to supply pressure input signals for both 1-PIC-47-189 and 1-PIC-47-193) will provide the advantage of smooth control and significantly decrease the potential of the controllers positioning the valves in such a manner that they are producing opposing effects in the gland sealing steam system. The continuous control advantages of the single sense point outweigh the disadvantage of the infrequent transmitter failure. The control loops are being designed such that failures will result in the same valve failure states as loss of air to the valves.

The components involved are not included in the Technical Specifications, and, therefore, do not have any effect on the margin of safety as defined in the basis for any Technical Specification.

Although the rerouting of the discharge line from the LP Heaters to the Main Condenser requires a change to a drawing in the FSAR, the change is not safety related and does not have specific design, operational, or performance requirements specified in the FSAR or Technical Specifications.

SA-SE Number *WBPLMN-97-065-0*

Affected Documents	*Document Type*	*Safety Assessment Title*	*Implementation Date:*
DCN W-39484-A	DCN	DCN making TACF 0-96-54-77 permanent connecting primary makeup water system to the discharge piping for both CDCT pumps and monitor tank pumps	11/12/97

Description of Change, Test, or Experiments

After a radioactive liquid release from either the Cask Decontamination Collector Tank (CDCT) or the Monitor Tank pump, a flush is required to allow the background setpoint for the release monitor (O-RE-90-122) to remain below the maximum allowed by Plant procedures. Temporary Alteration Control Form (TACF) 0-96-54-77 allowed a temporary check and manual isolation valves to be installed at the CDCT filter (O-FLTR-77-IA) vent and the Monitor Tank pump (O-PMP-77-2906) casing drain piping connections. The CDCT vent pipe cap and the Monitor Tank pump casing drain blind flange were temporarily removed to facilitate connection of the valves. Installation of these valves will allow the process piping from the CDCT and Monitor Tank through the discharge liquid radwaste release monitor to be flushed with clean water after each liquid release.

Design Change Notice (DCN) NO. 39484-A is making the TACF permanent. This DCN connects Primary Makeup Water System (81) to the discharge piping for both the CDCT pumps and Monitor Tank pumps. Each connection contains two isolation valves, a check valve, and associated piping. These changes will insure that the applicable piping and radiation monitor are sufficiently flushed to remove any residual radioactivity.

Safety Evaluation Summary

These system's associated components, piping, and valves are located in the Auxiliary Building on elevation 692. This equipment does not perform a primary safety function (except for containment isolation which has not been affected), are installed in a Seismic Category I structure, and are not used during any accident. The Chapter 15 accident analysis identifies an unexpected and uncontrolled release of radioactive xenon and krypton fission product gases stored in a Waste Gas Decay Tank as a consequence of a failure of a single Waste Gas Decay Tank or associated piping. This modification and equipment is not associated with the accident described above, does not increase the consequences of an accident previously evaluated, and is bounded by the existing analysis. This DCN does not change the logic or function of any system that is important to safety. A review of the detailed changes leads to the SE conclusions that this change is safe and does not constitute an USQ.

FSAR Section 11.2.4 identifies equipment faults which could occur with moderate frequency, including fuel cladding defects in combination with malfunctions in the Liquid Radwaste Processing System such as pump or valve failures or evaporator failures. This DCN does add new radwaste manual isolation and check valves. The new check valves have a potential to fail in the open position but if they do fail open, cross contamination would still be prevented since the radwaste isolation valves are normally closed. Also, this change is not associated with the protective features used to detect and mitigate the effects of these events. The equipment involved in the modification does not interface with any equipment whose malfunction could result in an accident which has been evaluated in the FSAR. This DCN does not change or affect the design basis for any system that is important to safety.

These changes do not affect any equipment required for safe operation or shutdown. In the event of a DBA, all safety related equipment is expected to operate as designed to limit the consequences of the DBA. This DCN adds piping, check and isolation valves to connect Primary Makeup Water System (81) to the discharge piping for both the CDCT pumps and Monitor Tank pumps. The previously evaluated malfunctions of radwaste components were reviewed and there is no increase of the consequences of these malfunctions. This change does not result in a radioactive release in excess of those established by 10 CFR 20 and 10 CFR 100 since this DCN does not create a new radioactive liquid or gaseous effluent release pathway as defined in ODCM. No new potential single failures of existing components will occur as a result of adding piping, check and isolation valves to connect Primary Makeup Water System (81) to the discharge piping for both the CDCT pumps and Monitor Tank pumps. Neither will this change cause this system or any system important to safety to fail to fulfill its functional requirements. These system's associated components, and piping do not perform any accident mitigation function except for containment isolation valves which have not been affected. The accidents and/or malfunctions associated with the Radwaste system is a failure of Waste Gas Decay Tank or associated piping and failure of radwaste components. Although this change does affect radwaste components, this equipment is not used in the mitigation of these accident/malfunctions and does not change the radiological consequences of an accident previously evaluated in the FSAR. A malfunction of the

Radwaste check valves to the open position would not prevent operations from flushing the release piping since, as an alternate, the flush piping could still be isolated with the radwaste isolation valves and a temporary connection added to the system similar to the TACF.

These changes do not reduce the margin of safety identified in the applicable Technical Specifications. These changes do not prevent any component from performing its function as described in the Technical Specifications. The ODCM limits for releases from either the Liquid Radwaste Processing System or the Gaseous Waste System are not revised or challenged by these changes.

SA-SE Number *WBPLMN-97-067-0*

Affected Documents	*Document Type*	*Safety Assessment Title*	*Implementation Date:*
FSAR Change Pkg. 1486 WBPER970262 TRM B3.3.5 Rev. 6	FSAR	Hi-Pressure Main Turbine Steam Chests	9/8/97

Description of Change, Test, or Experiments

WBPER970262 was written to revise and clarify FSAR Section 10.2.4 which, in its current form, has been incorrectly interpreted to indicate that the hi-pressure main turbine steam chests (2) are partitioned such that one throttle valve is in series with a governor valve. This is incorrect, the steam chests are not pardoned. The steam chests are large open areas (common to both throttle valves and governor valves). The steam flows into the steam chest via the throttle valve and exits the steam chest through one or both of the governor valves. The HP turbine and associated throttle and governor valves are not safety related and are not required to perform a primary or secondary safety function. The closure time of 0.15 seconds remains unchanged for these valves. Other physical attributes for the throttle and governor valves remains unchanged. All turbogenerator protective trips that will automatically trip the turbine due to turbine (mechanical) and generator (electrical) abnormalities remain unchanged. There are no plant modifications created by this FSAR change.

The Watts Bar Safety Evaluation Report (SER) and Supplements 1-20 have been evaluated for impacts. Specifically, the SER (sections 10.2 and 10.3) do not address the arrangement of the HP turbine throttle and governor valves in the steam chests. This documentation change only to the FSAR does not impact the NRC's understanding of the design and operation of Watts Bar Nuclear Plant as described in the FSAR because the function of the HP turbine throttle and governor valves remains unchanged. There are no plant modifications created by FSAR Change Package #1486.

Safety Evaluation Summary

Implementation of this FSAR change package does not affect any FSAR Chapter 15 fault or operational transient evaluations. Nor does it increase the probability of occurrence or consequences of any analyzed event. The existing Chapter 15 safety analyses assumes the closure of the throttle and governor valves for those events requiring turbine trip. However, this is not a safety function. There are no WBN design basis events for which the Turbine Generator Control and Protection System (TGCPS) is required to operate. The major plant safety concern for the TGCPS is the prevention of generation of turbine missiles due to a turbine overspeed condition (uncontrolled run away of the turbine). The possibility for an accident or malfunction of a different type than any previously evaluated in the FSAR remains unchanged. This FSAR change does not alter the existing safety analyses. No Tech Spec margins of safety are reduced. This change does not adversely affect any equipment important to safety, either directly or indirectly. Therefore, on the basis of the evaluation of effects, it is concluded that implementation of this FSAR Change Package #1486 is acceptable from a nuclear safety standpoint and no unreviewed safety question exists.

SA-SE Number *WBPLMN-97-073-0*

Affected Documents	*Document Type*	*Safety Assessment Title*	*Implementation Date:*
FP-04, -05, -07	Fire Protection Program	Fire Protection Report - Revision 10 Changes	2/6/98

Description of Change, Test, or Experiments

Fire Protection Report revision 10 incorporates the changes from change packages FP-04, -05, and -07 and documents the latest change to the combustible load summary into Table 1-1. The changes to Section II of the FPR include minor editorial/administrative clarifications, organizational title changes, clarification of equipment operation requirements, fire watch criteria, alternative temperature monitoring methods, deletion from Table 14.8-1 of fire doors in non-regulatory required barriers and changed rating of one door, and added ability to perform Technical Operability Evaluations. The addition of the Aux. Control Air System was covered by change package FP-04 (SA/SE WBPLEE-96-029-0). The thermal overloads for active valves were added to Table 14.10 to provide additional information on their fire protection requirement. The addition of the Positive Displacement Charging Pump to Section III was for clarification of why the pump is in the FPR. AFW Pump Recirculation Control valves are covered by change package FP-05 (SA/SE WBPLMN-96-022-0). Section V changes included organizational title changes. The changes in Section VI were to correct the type of detectors in Room 729.0-A5 and correct the required fire rating of door A122. Manual Operator action requirements changes are covered by change package FP-07 (SA/SE WBPLEE-96-084-0). Section VII changes included evaluations to justify reducing the surveillance frequencies for fire protection features in the Reactor Building Equipment Hatch (757.0-A11), eliminate testing and surveillance of fire dampers in the HVAC ducts for the Spent Fuel Pool and Fuel Transfer Canal, and evaluation of fire door W9 in the Intake Pumping Station. This revision also converts the FPR from Word Perfect to Word. This resulted in a complete replacement of all text and tables in the FPR. This revision of the FPR does not change the conclusions of NRC's Safety Evaluation Report as documented in NUREG-0847, Supplements 18 and 19.

Safety Evaluation Summary

1. Minor editorial/administrative clarifications and correcting the TVA organizational titles in the Fire Protection Report (FPR) to the current names has no affect on nuclear safety. This is an administrative change only.
2. Clarification of the equipment operating requirements to satisfy Appendix R requirements ensures that the equipment will be available to perform its intended function in the event of an Appendix R fire thereby ensuring nuclear safety.
3. Changes the fire watch requirements for areas when the suppression system is out of service and redundant systems can be affected by a single fire. The fire watch for an inoperable sprinkler system with the associated detection still operable was changed from continuous to hourly roving. This is reasonable since the early warning detection placed through out the affected area is still operable to provide the early warning annunciation to a constantly attended location. This early warning will ensure that the manual fire fighting capability (e.g., fire brigade), which ensures the fire is extinguished, is activated in a timely manor. The hourly fire watch will ensure potentially hazardous conditions are addressed before they are allowed to become a threat to the plant.
4. The Reactor Building has always been considered inaccessible for a fire watch at power operation due to radiological concerns. The shutdown of the unit for the loss of fire detection was not considered reasonable and this was recognized in the standard Technical Specifications. Thus the standard Technical Specifications provided an alternate means to a fire watch to monitor for fire upon the loss of detection in this area. The FPR has utilized this alternate monitoring means and this FPR revision clarifies how this monitoring is to be done. The specified means utilizes Technical Specifications instrumentation thus the maintenance and testing of the instrumentation to ensure accuracy is provided.
5. The list of fire doors in the FPR contains the doors that are required to comply with Appendix R and Appendix A to Branch Technical Position 9.5-1. Other doors are maintained as fire doors, but are not required to ensure safe shutdown in the event of a fire. Alternates to Appendix A separation requirements have been provided in the past and accepted by the NRC as documented in the SER, supplements 18 and 19. The doors being removed from the list are a part of one of these alternates to Appendix A and were noted on the compartmentation drawings as not required for Appendix R nor A. The removal of these doors from the list makes the compartmentation drawings and the list match.
6. The fire rating requirement of fire door A122 is being changed to reflect the correct rating. The door was shown as a 3-hour door and it only needs to be qualified for installation in a 2-hour barrier.
7. Section 14.10 deals with plant process equipment that is a part of the fire safe shutdown equipment (FSSD) population and was not in the Technical Specification, Technical Requirement, etc. As such, there is no clear NRC or standard guidance on how this equipment is to be addressed should it be inoperable. Thus generic requirements have been provided in Section 14.10. In some cases plant shutdown is called for if the equipment is not operable which may be excessive for the situation. To allow for equipment alternate compensatory measures that maintain the level of safety, Section 14.10 is being revised to address use of TACF process to provide alternate compensatory measures. This TACF process will allow for the plant to take credit for the redundancy built in the plant design thus using other equipment to achieve the same level of safety and ensure the same level of review is afforded to the change.

8. Providing additional information to clarify the fire protection requirements of active valves with thermal overloads provides additional confidence that the fire protection requirements of these valves are being maintained.
9. The Positive Displacement Charging Pump is not required for fire safe shutdown. This change provides a note to explain that the PD pump is not an active component and must not spuriously operate.
10. The type of detectors located in the Cask Loading area (room 729-A5) was incorrect in the FPR. This change corrects that typo to show the correct type detector. There was no physical change.
11. The FPR documentation for the outage test frequency of the RB Equipment Hatch Room (757.0-A11) is being added. This is acceptable because the room is not accessible during operation and because there are no fire hazards in the room during operations. In addition the suppression system piping is air supervised such that a break in the piping would be detected and send an annunciation to the Main Control Room central fire alarm station. The fire detection system is also supervised such that trouble on the fire detection system would also generate an annunciation at the central alarm station.
12. The fire dampers in the ducts to the Spent Fuel Pool and the Fuel Transfer Canal are not needed to prevent a fire from propagating through the ducts. The testing and surveillance requirements were to ensure that the dampers would close in the event of a fire. Since they are not needed, the need to ensure they will close is no longer needed.
13. Door W9 in the Intake Pumping Station has a gap that exceeds the maximum permitted by G-73. The evaluation determined that the door still provides the required fire resistance. This change to the FPR documents that evaluation in the deviation section of the FPR.
14. The duct detectors description in Part II, Section 12 was revised to reference the use of smoke type detectors in the duct detector assemblies instead of specifying the operating principle (i.e., ionization or photoelectric). This change still addresses the general type detector used in the plant and is not a change to the plant as designed and licensed. This change reduces excess detail and will allow for plant changes as needed as detection equipment is improved and incorporated into the design.
15. This change provides a bases for OR-14.1.4. This OR has already been incorporated into the FPR by an earlier revision and the reasons for the OR are being provided to clarify the intent.
16. The Section 12.10.4, Fire Doors, was clarified to match plant practice on how these doors are repaired. This practice ensures that proper reviews, in accordance with upper tier procedures, are performed for repairs that are other than a like-for-like item replacement.
17. Time studies were conducted before WBN was licensed to allow a continuous fire watch to cover more that one fire area at a time without putting undue physical stress on the fire watch. These time studies were made of selected areas where specified fire protection equipment outages affected multiple fire areas in close proximity. This addressed a concern of the NRC that the fire watch may be assigned too many areas. Additional equipment outages have since been determined to have similar physical aspects as the first set of fire protection equipment outages. Additional time studies have been conducted using the same criteria to allow a continuous fire watch cover multiple fire areas and the results are being added to this revision.
18. This change clarifies that the FPR does not have any requirements similar to the Technical Specification of 3.0.3, with the exception of certain conditions associated with Section 14.10, nor 3.04. Fire protection requirements similar to those found in OR-14.1 through 14.9 are provided in the standard Technical Specifications. The standard Technical Specifications specifically state that the provisions of LCO 3.0.3 and 3.0.4 do not apply to fire protection equipment and so this is being added as clarification and is consistent with the standard Technical Specifications. Section OR-14.10 does not have a similar section in the standard Technical Specifications so judgment was used to determine the appropriate action statements. This equipment is not Technical Specification equipment but is needed for fire safe shutdown. The shutdown statement of OR-14.10 is felt to be effective enough to endure equipment is maintained operable to the maximum extent possible and thus an additional mode change restriction statement is not needed.

19. The testing frequency of the detection for the sprinklers protecting the Containment Purge Air Filter Units is being changed from every 6 months to every refueling outage. The operability requirement for these detectors is that they be operable when the equipment they protect is required to be operable. The Containment Purge Air Filter Units are only required to be operable during irradiated fuel movement during Mode 6. To test these detectors requires the operation of the units during normal operation and this places the unit in an abnormal condition and causes multiple LCO entries. This testing frequency change allows for the testing of the detectors before the associated units are needed but prevents having to declare the detectors inoperable, and instituting the associated administrative paperwork, between fuel moving events.
20. The test frequency for cycling testable valves in the water supply in inaccessible areas, TIR-14.2.e.d, has been changed from 18 months to once each refueling outage. This is for consistency with the normal access frequency to inaccessible areas which is during refueling outages. At this time this TIR change does not affect any equipment as System 26 is presently installed and as the plant operating conditions currently permit.
21. The two duct detectors for the Unit 2 Post Accident Sampling Facility (PASF) filter housing have been removed from Table 14.1 and thus are no longer to be tested to meet FPR requirements. Duct detectors require air movement to operate properly. These two duct detectors are in a section of Unit 2 PASF duct that is interfaced out and blanked off so the duct detectors are useless. The duct detectors will probably be interfaced out at a later time. The removal of these two duct detectors from this testing requirement does not affect the ability of System 13 to detect fires.

For the reasons stated above, these changes do not constitute a USQ.

SA-SE Number *WBPLMN-97-101-1*

Affected Documents	*Document Type*	*Safety Assessment Title*	*Implementation Date:*
DCN M-39592-A FSAR Change Pkg. 1492	DCN	Revise Setpoints to Auto Start and Stop, Revise Associated Alarm Setpoints, and the Setpoint for Pressure Control Valve on the Inlet to the WGCs	10/5/97

Description of Change, Test, or Experiments

The present auto start/stop setpoints for the Waste Gas Compressors (WGCs) is 2.0 and 0.5 psig respectively. Before the WGCs auto stop, the following can occur: (1) when processing to the Mobile Demineralizers from the Chemical Volume and Control System (COCS) Holdup Tanks (HUTs), if a WGC auto starts the nitrogen (N2) gas feed to the HUTs starts opening at .7 psig decreasing which places excess gas to the "in-service" Waste Gas Decay Tank (WGDT) prior to WGC auto stop, (2) the HUT cover gas starts opening at 1.5 psig decreasing which places excess gas from the "cover" WGDT to the "in-service" WGDT prior to WGC auto stop, (1) transfer of any gas from the "covers to the "in-service" WGDT starts a new 60 day release clock to be generated, prolonging the length of time to hold waste gas prior to release, and (4) if the HUT recirculation pump is operating when a WGC auto starts the recirculation pump will automatically shutdown at 1.Q psig decreasing, prior to WGC auto stop.

To prevent the previous observed problems, Design Change Notice (DCN) NO. 39592-A revises the setpoint for 0-PS-77-88 to auto start at 3.5 psig and auto stop at 2.0 psig signals instead of the present 2.0 psig and 0.5 psig respectively. The DCN also revises the associated alarm setpoints, the setpoint for the pressure control valve (0-PCV-77-89) on the inlet to the WGCs from 1.5 to 3.0 psig, the FSAR, and the System Description N3-77A-4001 to reflect the new operating pressures. This change should decreased the current usage of N2 and cover gas.

Safety Evaluation Summary

These systems' associated components, piping, and valves are located in the Auxiliary Building on elevation 713. This equipment does not perform a primary safety function (except for containment isolation which has not been affected), are installed in a Seismic Category I structure, and are not used during any accident. The Chapter 15 accident analysis identifies an unexpected and uncontrolled release of radioactive xenon and krypton fission product gases stored in a Waste Gas Decay Tank as a consequence of a failure of a single Waste Gas Decay Tank or associated piping. This change and equipment, although apart of the Gaseous Waste Disposal System, is not associated with the accident described above, does not increase the consequences of an accident previously evaluated, and is bounded by the existing analysis. This DCN does not change the logic or function of any system that is important to safety. A review of the detailed changes leads to the SE conclusions that this change is safe and does not constitute an USQ.

FSAR Section 11.2.4 identifies equipment faults which could occur with moderate frequency, including fuel cladding defects in combination with malfunctions in the Liquid Radwaste Processing System such as pump or valve failures or evaporator failures. This DCN revises the setpoint for WGCs to auto start and auto stop, revises the associated alarm setpoints, and revises the setpoint for 0-PCV-77-89. Also, this change is not associated with the protective features used to detect and mitigate the effects of these events. The equipment involved in the change does not interface with any equipment whose malfunction could result in an accident which has been evaluated in the FSAR. This DCN does not change or affect the design basis for any system that is important to safety.

These changes do not affect any equipment required for safe operation or shutdown. In the event of a DBA, all safety related equipment is expected to operate as designed to limit the consequences of the DBA. This DCN revises the setpoint for Waste Gas Compressors (WGCs) to auto start at 3.5 psig and auto stop at 2.0 psig signals instead of the present 2.0 psig and 0.5 psig respectively, revises the associated alarm setpoints, and revises the setpoint for the pressure control valve (0-PCV-77-89) on the inlet to the WGCs from 1.5 to 3.0 psig. The previously evaluated malfunctions of Radwaste components were reviewed and there is no increase of the consequences of these malfunctions. This change does not result in a radioactive release in excess of those established by 10 CFR 20 and 10 CFR 100 since this DCN does not create a new radioactive gaseous effluent release pathway as defined in ODCM. No new potential single failures of existing components will occur as a result of revising the setpoint for WGCs to auto start and auto stop, the associated alarm setpoints, and the setpoint for 0-PCV-77-89. Neither will this change cause this system or any system important to safety to fail to fulfill its functional requirements. These systems' associated components and piping do not perform any accident mitigation function except for containment isolation valves which have not been affected. The accidents and/or malfunctions associated with the Radwaste system is a failure of Waste Gas Decay Tank (FSAR Section 15.3.5) or associated piping and failure of Radwaste components.

Although this change does affect Radwaste components, this equipment is not used in the mitigation of these accident/malfunctions and does not change the radiological consequences of an accident previously evaluated in the FSAR.

These changes do not reduce the margin of safety identified in the applicable Technical Specifications. These changes do not prevent any component from performing its function as described in the Technical Specifications. The ODCM limits for releases from the Gaseous Waste Disposal System are not revised or challenged by these changes.

SA-SE Number *WBPLMN-97-102-0*

Affected Documents

Special Test STI-97-01 Rev. 0

Document Type

Special Test

Safety Assessment Title

Revised Outside Air Temperature Limits for Diesel Generator Room Exhaust Fan

Implementation Date:

9/22/97

Description of Change, Test, or Experiments

Calculations EPM-AMP-081789, revision 6 (RIMS B26 970728 300) and EPM-CES-082989, revision 6 (RIMS B26 970728 301) established revised outside air temperature limits for allowing one Diesel Generator (DG) room exhaust fan, or the generator and electrical panels ventilation supply fan to be out-of-service while still maintaining acceptable temperatures within the DG room. These revisions resulted in a reduction in the temperatures from the values previously listed in the DG System Operating Instructions (SOIs). In addition, Essential Raw Cooling Water (ERCW) temperature was included in the criterion since it affects the amount of heat removed by each DG heat exchanger. Calculation EPM-JJL-070789, revision 4 (reference 2.4) will be used to establish an outside air temperature limit allowing for one 480 Volt Electrical Board Room exhaust fan to ventilate two adjacent board rooms of a common train. This outside temperature limit had not previously defined.

Air flow through each DG room is induced by two exhaust fans which draw outside air into the air intake room, through the DG room and into the air exhaust room. The generator and electrical panels ventilation supply fan draws air from the air intake opening (in the DG room) and ducts this air directly to both DG electrical panels and near the generator air intakes. In doing so, it provides location specific (spot) cooling to critical components within the DG room. Each 480 Volt Electrical Board Room is continuously ventilated by an exhaust fan. Air enters the board room through a roof mounted intake vent, flows through the room before being drawn into exhaust ductwork leading to the exhaust fan located in the exhaust fan room.

DG room air temperatures, both the bulk air temperature and temperatures at specific locations within the room, are a function of the heat loads present, the mass flow rate of air passing through the room, and the entering air temperature. Major sources of heat within each DG room are the diesel engines and electrical generator. The generator heat load contribution is a function of how it is loaded electrically and is controlled throughout each test. The diesel engine heat load contribution to the room air is a function of its surface temperature, surface area and convective heat transfer coefficient. Surface temperature is controlled by the jacket water cooling system, while the heat transfer coefficient, in simplified form, is a function of the air velocity and surface geometry. Jacket water cooling system performance is affected by ERCW entering temperature, ERCW flow rate, and heat exchanger. In consideration of all of these variables, it is not practical to calculate location specific temperatures within the room, or bulk air temperatures for postulated worst case accident conditions (reduced ERCW flow rate, maximum heat exchanger fouling factor, changing heat transfer coefficients, etc.). The most accurate method for eliminating these unknowns is to manually increase the diesel engine jacket water temperature to near its high temperature alarm limit (187°F), de-energize one DG room exhaust fan, or the generator and electrical panels ventilation supply fan and directly measure temperatures at specific locations within the DG room. Thus, the purpose for this special test is to simulate worst case conditions with either one DG room exhaust fan off, the generator and electrical panels ventilation supply fan off, or all fans running normally. The outside air temperature recorded during the test can be extrapolated by a subsequent design calculation to establish accurate limiting outside air temperature values for each fan out-of-service condition. The 480 volt electrical board room data recorded during the test can be used to establish the limiting

Safety Evaluation Summary

The special test of DG 2A-A does not affect the FSAR chapter 15 Design Basis Accident (DBA) evaluations. Critical temperatures are monitored during the test and caution statements are included to re-establish full ERCW flow to each diesel engine heat exchanger and (or) re-establish full ventilation flow if maximum temperatures are reached. Maximum temperatures for the diesel and generator were chosen below those values defined in the DG 2A-A System Operating Instruction SOI-82.03. The maximum temperature for the bulk room air is the abnormal maximum as defined on the DG building environmental data drawing. The maximum electrical panels temperature is based on the diesel vendor's recommendation and is equal to the abnormal maximum for the room bulk air. These precautions will support continued operation of DG 2A-A during its 24 hour load test (O-SI-82.5). However, if DG 2A-A should be declared inoperable during, or prior to the test the probability of occurrence of an accident previously evaluated 1n the FSAR will not be increased. This is based on Technical Specification 3.8.2 which is applicable since the plant will be in mode 5 (cold shutdown) or 6 (refueling) prior to and during the test. Technical Specification 3.8.8 "AC Sources - Shutdown" requires "One qualified circuit between the offsite transmission network and the onsite Class 1E AC electrical power distribution subsystem(s) required by LCO 3.8.10, Distribution Systems - Shutdown; and two diesel generators either train A or train B capable of supplying one train of onsite Class 1E AC electrical power distribution subsystem(s) required by LCO 3.8.10." When the plant is shut down, the Technical Specification requirements ensure that the plant has the capability to mitigate the consequences of postulated accidents. However, assuming a single failure and concurrent loss of all offsite or all onsite power is not required (reference Technical Specification 3.8.2, BASES). The rational for this is based on the fact that many DBAs that are analyzed in MODES 1,2,3 and 4 have no specific analyses in mode 5 and 6. Worst case bounding events are deemed not credible in modes 5 and 6 because the energy contained within the reactor pressure boundary, reactor coolant temperature and pressure, and the corresponding stresses result in the probabilities of occurrence being significantly reduced or eliminated, and of min1mal consequences. These deviations from DBA analysis assumptions and design requirements during shutdown are allowed by the LCO for required systems. The operability of the minimum AC sources during modes 5 and 6 and during movement of irradiated fuel assemblies ensures that: a) The plant can be maintained in the shutdown or refueling condition for extended periods; b) sufficient instrumentation and control capability is available for monitoring and maintaining the plant status; and c) adequate AC electrical power is provided to mitigate events postulated during shutdown, such as a fuel handling accident. The critical temperatures monitored during the test and associated caution statements will assure that the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR 1S not increased. The special test will not increase the consequences of an accident, or increase the consequences of a malfunction of equipment important to safety previously evaluated in the SAR based on the Technical Specification 3.8.2 requirements discussed above. Similarly, the special test will not create a possibility for an accident or malfunction of a different type than any previously evaluated in the SAR since the plant will be operating within the requirements of Technical Specification 3.8.2. The safety margins for AC sources during shutdown addressed in Technical Specification 3.8.2 are not reduced even if DG 2A-A were declared inoperable. Therefore, it is concluded that Special Test STI-97-01 are outside air temperature when this fan is an out-of-service condition.

The special test will be performed during the 24 hour load test portion of surveillance instruction 0-SI-82-5, "18 Month Loss of Offsite Power - DG 2A-A". During the initial 2 hours of the test, while the DG is loaded to >4620 kW and <4840 kW, data will be collected in 480 Volt Board Room 2A with the associated ventilation fan operating normally. Air inlet and outlet dry bulb temperatures as well as barometric pressure, inlet wet bulb temperature, and surrounding room temperatures will be measured and recorded. After the initial 2 hours and once generator stator winding temperatures have stabilized, the generator and electrical panels ventilation supply fan 2-FAN-30-492 will be de-energized (simulating an out-of-service condition for this fan). ERCW valves associated with heat exchangers 2AI and 2A2 will be throttled from their balanced position until diesel water jacket temperatures of 185+/-2°F are obtained (simulating reduced ERCW flow and a maximum fouling condition which could exist during worst case accident conditions). Jacket water return temperature to each heat exchanger will be recorded and monitored throughout the test. Enough data will be collected to demonstrate that thermal equilibrium has been achieved over a period of approximately 3 hours. Air inlet and outlet dry bulb temperatures for the DG room as well as barometric pressure, inlet wet bulb temperature, and surrounding room temperatures will be measured and recorded. Air inlet temperatures and velocities to the generator air intakes (6 total) and the internal temperatures of the DG electrical panels will be recorded generator air intakes (6 total) and the internal temperatures of the DG electrical panels will be recorded and monitored throughout the test. Maximum temperature limits for the generator stator winding, bearings and within each electrical panel will not be exceeded. DG room bulk air temperature will be determined based on a weighted average of the exhaust air temperature from the DG room. This temperature will not be allowed to exceed the abnormal maximum value of 120°F. If any of the maximum temperatures are reached during the test, steps will be taken to restore the generator and electrical panels ventilation supply fan, both DG room exhaust fans, or increase ERCW flow to the heat exchangers. Diesel jacket water cooler performance data will be collected concurrently to determine the total heat load removed by each heat exchanger and to record jacket water temperatures. At the end of this portion of the test, the generator and electrical panels ventilation supply fan will be restarted.

The next phase of the test will require that DG room exhaust fan 2-FAN-30-452 be de-energized (simulating an out-of-service condition for the exhaust fan with the greatest flow rate). ERCW valves will remain in a throttled position such that jacket water temperatures are maintained at 185+/-2°F. Data will be collected over a period of approximately 3 hours to demonstrate thermal equilibrium has been achieved. The same data will be collected as during the simulated electrical panels ventilation supply fan out-of-service condition discussed previously. The same critical temperatures will be monitored and the DG room exhaust fan will be restarted if required. At the end of this portion of the test, DG room exhaust fan 2-FAN-30-452 will be restarted.

The final phase of data collection will be performed with all fans (both DG room exhaust fans and the generator and electrical panels ventilation supply fan) running normally. Again, data will be collected over a period of approximately 3 hours to demonstrate that thermal equilibrium has been achieved. ERCW flow to each heat exchanger will remain throttled to maintain DG jacket water temperature at 185+/-2°F. The same critical temperatures will be monitored as discussed above. At the end of this final phase, ERCW throttling valves will be returned to their original balanced position per TI-31.08 and the DG 2A-A 24 hour load test will continue as normal.

acceptable from a nuclear safety standpoint and no unreviewed safety question exists.

SA-SE Number *WBPLMN-97-117-0*

Affected Documents	*Document Type*	*Safety Assessment Title*	*Implementation Date:*
TI-68.012	Procedure	Installation of the Mansell Level Monitoring System (MLMS) and the Performance of a Vacuum Refill of the Reactor Coolant System (RCS)	9/6/97

Description of Change, Test, or Experiments

The activities proposed and covered by this Safety Assessment/Safety Evaluation consist of installation and operation of the Mansell Level Monitoring System (MLMS), and Vacuum Refill of the RCS.

During maintenance and refueling outages it is necessary to reduce Reactor Coolant System pressure to atmospheric to allow opening of the Reactor Coolant System (RCS) for maintenance and fuel movement. In some cases it may be necessary to reduce the water level in the RCS to a level where maintenance can be performed. During periods of reduced inventory, it is vital that level measurement systems be in service which will rapidly provide the operators with an accurate and reliable level indication. The MLMS will provide an additional means of shutdown, depressurized RCS level indication.

The MLMS utilizes redundant digital quartz pressure transducers mounted at a reference location (pressurizer vent and reactor head vent) and at a liquid head location (loop #1 crossover leg drain line). The pressure information from these points are transmitted to the Control Room, processed by a computer dedicated to the MLMS., and displayed to the control room operators in FEET and INCHES to a resolution of 0.1 inches of water. The display system also provides an operator interface where the required information can be entered to set the proper zero range, level alarm setpoints, temperature, and boron concentration.

After maintenance and refueling, a method to remove the air and non-condensable gasses from the RCS must be used. Currently, Watts Bar procedures utilize the "Sweep and Vent" evolution to remove air and non-condensables. This procedure requires several reactor coolant pump (RCP) "bumps" (starts and stops), as well as several pressurizing/depressurizing evolution's. The "sweep and vent" process is time consuming, causes increased radiation exposure to personnel performing the venting operation, and the numerous starts and stops of the RCPs cause increased wear to the RCP motor and seal systems. An alternate means of air and non-condensable gas removal is to remove the air and gasses prior to refill by exposing the RCS to a vacuum. The Reactor Coolant Vacuum Refill System (RCS Vacuum Refill) is designed to remove air and non-condensable gasses from the RCS in Mode 5 following refueling operations during plant startup. The gases will be drawn from the RCS, including the steam generator tubes, to the reactor vessel head and pressurizer. The RVHVS valves and pressurizer power operated relief valves (PORV) are opened to allow the gases and air to flow through the pressurizer relief tank (PRT) and the RCS Vacuum Refill suction header. The gases and condensed water are discharged to predetermined locations depending on the plants effluent and waste processing system availability. After the gases are removed, the vacuum is maintained until the RCS is filled. When the system is filled completely, the RCS is pressurized and the RCPs are started once.

Safety Evaluation Summary

The use of the MLMS and the RCS Vacuum Refill will not impair the safety function or performance of the reactor vessel internals, CRDMs, level monitoring systems, RCP seals, pressurizer, steam generators, tanks, pumps, valves, filters, demineralizers, heat exchangers or piping in any way. Procedural restrictions will be in place to isolate or recalibrate certain instruments with the potential for zero shifl when exposed to a negative pressure (vacuum). Note, the structural integrity of these instruments is not degraded as a result of being exposed to a vacuum. The use of the MLMS° and the RCS Vacuum Refill will not adversely affect the safe operation of Watts Bar Unit 1, and the use of the MLMS. and the RCS Vacuum Refill does not represent a potential unreviewed safety question.

SA-SE Number *WBPLMN-97-121-0*

Affected Documents	*Document Type*	*Safety Assessment Title*	*Implementation Date:*
DCN S-39707-A	DCN	System Description N3-77C-4001 Revised to Allow Use of a Coagulant to Remove Cobalt	10/14/97

Description of Change, Test, or Experiments

Flow from both the tritiated and non tritiated tanks is routed to a Mobile Demineralizer System (MDS). The processed water from the MDS is routed to either of two release tanks. The release tanks are either the Monitor Tank or the Cask Decontamination Collector Tank (CDCT). The contents of these tanks are discharged to the Cooling Tower Blowdown Down (CTB) or processed further, as necessary, to meet Offsite Dose Calculation Manual (ODCM) limits. The MDS removes most soluble and suspended radioactive materials from the waste stream via ion exchange and filtration. Cobalt 58 is present in the radwaste stream which requires treatment by the MDS prior to being processed to either release tank. Resins used in the MDS are very ineffective in removal of this activity. A pretreatment is required to enable the MDS to effectively remove the Cobalt 58.

Design Change Notice (DCN) NO. 39707-A revises the System Description N3-77C-4001, "Liquid Radwaste Processing System" to allow for the use of a coagulant to remove Cobalt 58. The coagulant (Nalcolyte 7134) will be injected into the MDS process stream to provide a chemical capture mechanism for Cobalt 58. The Nalcolyte will be mixed and injected as necessary in the piping for the MDS. The maximum concentration will be 40 parts per million (ppm). These changes will insure that the ODCM release requirements are satisfied for radioactivity.

Safety Evaluation Summary

These systems' associated components, piping, and valves are located in the Auxiliary Building on elevation 729. This equipment does not perform a primary safety function (except for containment isolation which has not been affected), are installed in a Seismic Category I structure, and are not used during any accident. The Chapter 15 accident analysis identifies an unexpected and uncontrolled release of radioactive xenon and krypton fission product gases stored in a Waste Gas Decay Tank as a consequence of a failure of a single Waste Gas Decay Tank or associated piping. This documentation change only is not associated with the accident described above, does not increase the consequences of an accident previously evaluated, and is bounded by the existing analysis. This DCN does not change the logic or function of any system that is important to safety. A review of the detailed changes leads to the SE conclusions that this change is safe and does not constitute an USQ.

FSAR Section 11.2.4 identifies equipment faults which could occur with moderate frequency, including fuel cladding defects in combination with malfunctions in the Liquid Radwaste Processing System such as pump or valve failures or evaporator failures. This DCN approves the use of a coagulant in the MDS, and does not add any new equipment since the MDS pump skid has the capability to inject this chemical. Also, this change is not associated with the protective features used to detect and mitigate the effects of these events. The equipment involved in the modification does not interface with any equipment whose malfunction could result in an accident which has been evaluated in the FSAR. This DCN does not change or affect the design basis for any system that is important to safety.

These changes do not affect any equipment required for safe operation or shutdown. In the event of a DBA, all safety related equipment is expected to operate as designed to limit the consequences of the DBA. This change approves the use of a coagulant in the MDS, and does not add any new equipment since the MDS pump skid has the capability to inject this chemical. The previously evaluated malfunctions of Radwaste components were reviewed and there is no increase of the consequences of these malfunctions. This change does not result in a radioactive release in excess of those established by 10 CFR 20 and 10 CFR 100 since this DCN does not create a new radioactive liquid or gaseous effluent release pathway as defined in ODCM. No new potential single failures of existing components will occur as a result of addition of this chemical. Neither will this change cause this system or any system important to safety to fail to fulfill its functional requirements. These systems' associated components, and piping do not perform any accident mitigation function except for containment isolation valves which have not been affected. The accidents and/or malfunctions associated with the Radwaste system is a failure of Waste Gas Decay Tank or associated piping and failure of Radwaste components. Although this change does affect radwaste components, this equipment is not used in the mitigation of these accident/malfunctions and does not change the radiological consequences of an accident previously evaluated in the FSAR.

These changes do not reduce the margin of safety identified in the applicable Technical Specifications. These changes do not prevent any component from performing its function as described in the Technical Specifications. The ODCM limits for releases from either the Liquid Radwaste Processing System or the Gaseous Waste System are not revised or challenged by these changes.

SA-SE Number *WBPLMN-97-142-0*

Affected Documents	*Document Type*	*Safety Assessment Title*	*Implementation Date:*
DCN S-39752-A	DCN	Essential Raw Cooling Water (ERCW) System Flow Diagram Revised per Drawing Deviation	1/20/95

Description of Change, Test, or Experiments

DCN S-39752-A revises Essential Raw Cooling Water (ERCW) system flow diagram 1-47W845-3 per Drawing Deviation (DD) 97-0155 to show the as-constructed location of the main header tap for 1 -TV-67-1213. Currently 1-47W845-3 shows the connection for 1-TV-67-1213 immediately downstream of where the 1/2-inch check valve bypass around 1-FCV-67-298-B connects to the main header. DCN S-39752-A will simply correct the depiction of 1-TV-67-1213 to show it connected immediately upstream of the 1/2-inch check valve bypass connection. FSAR Change Package 1500 changes Table 6.2.4-1, sheet 41 of 69 to make identical corrections to the detail for penetration X-72. These are minor documentation changes only to reflect the as-constructed configuration.

1-TV-67-1213 exists in the system for the single purpose of draining the system piping to support local leak rate testing of containment penetration X-72. Depiction of the as-constructed location for 1-TV-067-1213 by DCN S-39752-A does not impact the valve's function relative to containment penetration local leak rate testing. In addition, the change to show the as-constructed location of 1-TV-067-1213 on flow diagram 1-47W845-3 is supported by existing pipe stress analyses which are based on the as-constructed configuration control drawings rather than the flow diagram. Therefore ERCW, system performance and containment integrity are unaffected. The design basis and FSAR Chapter 15 accidents that require containment isolation and integrity have been reviewed and are unaffected by this minor documentation change. In addition, the response of the containment to design basis accidents as contained in FSAR Chapter 5 is unaffected. No physical changes are made by DCN S-39752-A. The change is documentation only to show the as-constructed location of the main header tap for 1-TV-67-1213. Therefore, there are no credible failure modes associated with the change. Tech Specs 3.7.8 and 3.6.3 have been reviewed with respect to the minor documentation change of DCN S-39752-A. The change to show the as-constructed location of 1-TV-67-1213 on the flow diagram has been determined to have no impact on the Tech Specs for either the ERCW system or the containment Isolation System

Safety Evaluation Summary

Based on the previous discussion, the minor documentation change to FSAR information incorporated by DCN S-39752-A does not impact the probability of occurrence or consequences of any accident or equipment malfunction currently evaluated in the FSAR. In addition, the documentation change does not create the possibility of an accident or equipment malfunction of a different type than previously evaluated and does not reduce the margin of safety as defined in the basis for any Technical Specification. Therefore, the changes of DCN S-39752-A do not constitute a USQ.

SA-SE Number *WBPLMN-97-145-0*

Affected Documents	*Document Type*	*Safety Assessment Title*	*Implementation Date:*
DCN S-39783-A	DCN	Revised Configuration Control Drawings to Address Drawing Deviations	1/21/98

Description of Change, Test, or Experiments

The equipment and instrument drains that drain to the Tritiated Drain Collector Tank (TDCT) are normally closed drains. These drains are normally designated with a "C" which stands for closed on the flow diagram. A drawing deviation (DD) was initiated to correct a drain on Configuration Control Drawing (CCD) 1-47W852-2 at "W" and "A5". This CCD is a Final Safety Analysis Report (FSAR) figure. Drains in the Hot Sample Room Unit 1 that are used for grab samples are currently shown on CCD 1-47W625-22 as teeing off downstream of the bypass valves. A DD was initiated to show that these drains tee off upstream of the bypass valves. This CCD is not a FSAR figure.

Design Change Notice (DCN) NO. 39783-A revises 1-47W852-2 to add a "C" for the closed drain, and revises 1-47W625-22 to move the tee upstream of the bypass valves

Safety Evaluation Summary

This system's associated components, piping, and valves are located in the Auxiliary Building on elevation 713. This equipment does not perform a primary safety function, are installed in a Seismic Category I structure, and are not used during any accident. The Chapter 15 accident analysis does not identify any failure that is associated with revising the CCDs. This change and equipment are not associated with increasing the consequences of an accident previously evaluated, and is bounded by the existing analysis. This DCN does not change the logic or function of any system that is important to safety. A review of the detailed changes leads to the SE conclusions that this change is safe and does not constitute an USQ.

FSAR does not identify any equipment faults which could occur as a result of this change. This documentation change only DCN revises the CCDs. Also, this change is not associated with the protective features used to detect and mitigate the effects of any events. The equipment involved in the change does not interface with any equipment whose malfunction could result in an accident which has been evaluated in the FSAR. This DCN does not change or affect the design basis for any system that is important to safety.

These changes do not affect any equipment required for safe operation or shutdown. In the event of a DBA, all safety related equipment is expected to operate as designed to limit the consequences of the DBA. This DCN revises the CCDs. No new potential single failures of existing components will occur as a result of this documentation change only DCN. Neither will this change cause this system or any system important to safety to fail to fulfill its functional requirements. These systems' associated components and piping do not perform any accident mitigation function. This equipment is not used in the mitigation of any accident/malfunctions and does not change the radiological consequences of an accident previously evaluated in the FSAR.

These changes do not reduce the margin of safety identified in the applicable Technical Specifications. These changes do not prevent any component from performing its function as described in the Technical Specifications.This system's associated components, piping, and valves are located in the Auxiliary Building on elevation 713. This equipment does not perform a primary safety function, are installed in a Seismic Category I structure, and are not used during any accident. The Chapter 15 accident analysis does not identify any failure that is associated with revising the CCDs. This change and equipment are not associated with increasing the consequences of an accident previously evaluated, and is bounded by the existing analysis. This DCN does not change the logic or function of any system that is important to safety. A review of the detailed changes leads to the SE conclusions that this change is safe and does not constitute an USQ.

FSAR does not identify any equipment faults which could occur as a result of this change. This documentation change only DCN revises the CCDs. Also, this change is not associated with the protective features used to detect and mitigate the effects of any events. The equipment involved in the change does not interface with any equipment whose malfunction could result in an accident which has been evaluated in the FSAR. This DCN does not change or affect the design basis for any system that is important to safety.

These changes do not affect any equipment required for safe operation or shutdown. In the event of a DBA, all safety related equipment is expected to operate as designed to limit the consequences of the DBA. This DCN revises the CCDs. No new potential single failures of existing components will occur as a result of this documentation change only DCN. Neither will this change cause this system or any system important to safety to fail to fulfill its functional requirements. These systems' associated components and piping do not perform any accident mitigation function. This equipment is not used in the mitigation of any accident/malfunctions and does not change the radiological consequences of an accident previously evaluated in the FSAR.

These changes do not reduce the margin of safety identified in the applicable Technical Specifications. These changes do not prevent any component from performing its function as described in the Technical Specifications.

SA-SE Number *WBPLMN-98-003-0*

Affected Documents	*Document Type*	*Safety Assessment Title*	*Implementation Date:*
FSAR Change Pkg. 1503	FSAR	Containment Spray Pump Net Positive Head (NPSH) Data Provided for the Containment Spray Pumps	1/29/98

Description of Change, Test, or Experiments

FSAR Section 9.2.7.1 states that ECCS and CSS Pumps NPSH data is tabulated in Table 9.2-3. The table tabulates NPSH data for the Residual Heat Removal (RHR) Pumps, the Centrifugal Charging (CC) Pumps, and the Safety Injection (SI) Pumps, but no NPSH data is tabulated for the Containment Spray (CS) Pumps. The NPSH data for the CS Pumps was originally contained in Table 9.2-3, but was inadvertently omitted by FSAR Change Package 0874 Supplement S1 which was performed during the review and,revision for the Mechanical/Nuclear Calculations Program. NPSH data for the CS Pumps is required to be included in the FSAR by NRC Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants." FSAR Change Package 0874 deleted the previously existing table and replaced it with a re-formatted table. The CS Pump NPSH data was inadvertently deleted by the SI calculation review team because their focus was on the SI System only, and it was not recognized that the previously existing table contained NPSH data for both the CS Pumps and the ECCS Pumps.

WBPER971427 identified the above problem, and FSAR Change Package 1503 is being issued as the Corrective Action to the PER. Existing design basis calculations provide the CS Pump NPSH data that will be included in Table 9.2-3 such that no new calculations are necessary.

FSAR Change Package 1503 replaces the NPSH data for the CS Pumps in Table 9.2-3. The data used is taken from revision 2 of TVA Calculation EP-RCP-120291, "Containment Spray Pump Net Positive Head (NPSH) Calculation. The design basis and FSAR Chapter 15 accidents that require containment isolation and integrity are unaffected by replacement of this data. In addition, the response of the containment to design basis accidents as contained in FSAR Chapter 5 is unaffected. No physical changes are made by FSAR Change Package 1503. The change is documentation only to replace information which was inadvertently omitted; therefore, there are no credible failure modes associated with the change. Tech Spec 3.6.6, Containment Spray System, has been reviewed with respect to the minor documentation change of this FSAR Change Package, and the change has been determined to have no impact.

Safety Evaluation Summary

Based on the previous discussion, the minor documentation change to FSAR information incorporated by FSAR Change Package 1503 does not impact the probability of occurrence or consequences of any accident or equipment malfunction currently evaluated in the FSAR. In addition, the documentation change does not create the possibility of an accident or equipment malfunction of a different type than previously evaluated and does not reduce the margin of safety as defined in the basis for any Technical Specification. Therefore, the changes of FSAR Change Package 1503 do not constitute a USQ.

SA-SE Number *WBPLMN-98-004-0*

Affected Documents

DCN S-39813-A

FSAR Change Pkg. 1505

Document Type

FSAR

Safety Assessment Title

Residual Heat Removal (RHR) System Operation During Refueling

Implementation Date:

2/3/98

Description of Change, Test, or Experiments

This safety evaluation addresses the scope of FSAR Change Package 1505 which clarifies section 5.5.7.2.2 for Residual Heat Removal (RHR) system operation during refueling. The text in this section is deleted that describes closing the isolation valves in the inlet line to the RHR loop for filling the refueling cavity. This action was only permitted prior to initial criticality and is no longer applicable. The change to the FSAR section 5.5.7.2.2 text is a clarification as the reformatted merits Tech Specs only allow the RHR loop to be removed from service prior to initial criticality. RHR system description N3-74-4001, section 3.1.2 is also revised by DCN S-39813-A to correct wording that is similar to that contained in the FSAR.

The design basis accidents evaluated in WB-DC-40.64 "Design Basis Events Design Criteria" and in FSAR Chapter 15 "Accident Analysis" have been reviewed and the condition 1 event (expected to occur frequently) of refueling and the condition IV, limiting faults, event of a fuel handling accident are applicable. However, this documentation change does not impact the results and conclusions of these analyses. Plant radioactive releases are unchanged and continue to remain well below the limits of 10 CFR 100.

There are no credible failure modes associated with this change. The change to FSAR section 5.5.7.2.2 complies with Tech Spec 3.9.6 which requires that two RHR loops be operable, and one RHR loop be in operation during low reactor vessel water level refueling operations. This limiting condition for operation assures removal of decay heat and mising of borated coolant to minimize the possiblity of criticality. RHR system operation is in compliance with Tech Spec 3.9.6 and according to SOI-74.01. The FSAR and design basis document simply require revision to achieve consistency with other documents.

Safety Evaluation Summary

The subject FSAR and system description changes do not constitute an unreviewed safety question because operation of the RHR system during low reactor vessel water level refueling operations is unchanged. The frequency in which this mode of operation must be entered is not increased and the probability of an RHR train malfunction while in the refueling mode is not increased. The radiological consequences of performing the refueling operation itself are unchanged. The radiological consequences of a malfunction of a train of RHR are not changed because operation of the RHR system is not changed. RHR system operation in accordance with Tech Spec 3.9.6 for refueling is unchanged such that the possibility for an accident or equipment malfunction of a different type than evaluated previously in the FSAR is not created. The subject FSAR and design basis document changes do not require operation of the RHR system in conflict with Tech Specs and no physical changes are performed such that the margins of safety as defined in Tech Spec bases are not reduced.

SA-SE Number *WBPLMN-98-008-0*

Affected Documents	*Document Type*	*Safety Assessment Title*	*Implementation Date:*
FSAR Change Pkg. 1504	FSAR	FSAR Revised to Clarify the Text to Better Represent the Current Operation of the Affected System	2/3/98

Description of Change, Test, or Experiments

The Final Safety Analysis Report (FSAR) is being revised to make clarifications to the text to better represent the current operation of the affected system. The table below list the affected FSAR sections and the reason for the revision.

9.3.2.2 This provides a clarification that the Waste Gas Analyzer samples are also taken in a hydrogen atmosphere.

9.3.2.3 This deletes the type of sample container that would be used during conditions approximating 1% failed fuel since the method for taking samples may vary depending on the condition. These samples are still required to be obtain during conditions approximating 1% failed fuel.

9.3.3.2.1 This deletes that samples are taken for Auxiliary Building Floor and Equipment Drain Sump (ABF&EDS) and deletes that the Additional Equipment Building Sump is pumped to the Tritiated Drain Collector Tank (TDCT) since these sumps are normally processed to the Floor Drain Collector Tank (FDCT) and samples are taken in the Waste Disposal release tanks prior to processing to the environs. These samples were not intended to be utilized for leakage detection since area monitor provide this function for equipment that is important to safety.

9.3.3.7 Demineralized water is used in Component Cooling System (CCS) which does not contain chromates, and CCS no longer uses chromates as a corrosion inhibitor. Thus the word chromated is being deleted as an editor change to make it consistence with the other sections of the FSAR.

9.3.4.1.4 &
9.3.4.2.2 This deleted that pH is specified in Table 5.2-10 and revises to refer to this table for the hydrogen concentration. The text should not have refer to the table since pH is not specified this table.

10.3.5.2 Ethanolamine (ETA) was added since the all volatile treatment (AVT) is ETA.

10.4.2.5 This section is being revised to clarify that only the low-range radiation monitor is used for Offsite Dose Calculation Manual (ODCM) for normal operation, and the high range monitor is employed for accidents. These monitors are used to indicate that a problem exist and initiate steps to determine the cause (based on the Abnormal Operating Instructions (AOI)). Based this investigation, the plant will then determine whether or not to shutdown. Thus this change deletes that the plant goes into a controlled shutdown as a result of these monitors since these monitors only indicate that a problem exist.

10.4.8.1 This revises that the ODCM controls the limits for releases.

11.2.3.1 This deletes that the FDCT is analyzed prior to further processing since after processing through the Mobile Waste Demineralizers, the release tanks are analyzed prior to being released.

Safety Evaluation Summary

The affected system's associated components, piping, and valves are located in the Auxiliary and Turbine Buildings. This equipment does not perform a primary safety function, and are not used during any accident except for containment isolation. The Chapter 15 accident analysis does not identify any failure that is associated with revising the FSAR. This documentation only change and equipment are not associated with increasing the consequences of an accident previously evaluated, and is bounded by the existing analysis. This revision to the FSAR does not change the logic or function of any system that is important to safety. A review of the detailed changes leads to the SE conclusions that this change is safe and does not constitute an USQ.

FSAR does not identify any equipment faults which could occur as a result of changes. Also, changes are not associated with the protective features used to detect and mitigate the effects of any events. The equipment involved in the change does not interface with any equipment whose malfunction could result in an accident which has been evaluated in the FSAR. This revision to the FSAR does not change or affect the design basis for any system that is important to safety.

These changes do not affect any equipment required for safe operation or shutdown. In the event of a DBA, all safety related equipment is expected to operate as designed to limit the consequences of the DBA. No new potential single failures of existing components will occur as a result of this documentation change only. Neither will this change cause this system or any system important to safety to fail to fulfill its functional requirements. These affected system's associated components, and piping do not perform any accident mitigation function. This equipment is not used in the mitigation of any accident/malfunctions and does not change the radiological consequences of an accident previously evaluated in the FSAR.

These changes do not reduce the margin of safety identified in the applicable Technical Specifications. These changes do not prevent any component from performing its function as described in the Technical Specifications.

This also revises that the ODCM controls the limits for releases.

12.2.1 This adds a clarification how the source terms were used.

This is a FSAR change only with no physical change to the facility.

SA-SE Number *WBPOSG-95-052-3*

Affected Documents	*Document Type*	*Safety Assessment Title*	*Implementation Date:*
DCN W-38238-B TS Bases Change Pkg. No. 95-090 Bases Revision 1	DCN	ESFAS Slave Relay Circuits	2/27/96

Description of Change, Test, or Experiments

DCN W-38238-B moves the ESFAS signals for 1-FCV-70-100-A and for the 6.9 KV Shutdown Boards Emergency Feeder Breakers (Diesel Generator Breakers) to other slave relays to allow testing which will not disrupt plant operation. A separate description of the change for each of these features is provided below. The DCN is a staged DCN. The stages are defined in the DCN and correspond to the completion of modifications for 1-FCV-70-100-A as one stage, and a stage for each Emergency Feeder Breaker for the 6.9 KV Shutdown Boards 1A-A, 1B-B, 2A-A, and 2B-B. This results in five stages in the DCN.

a. 1-FCV-70-100-A is one of four Component Cooling System to Reactor Coolant Pump (RCP) Oil Coolers Containment Isolation Valves which are designed to close on a Containment Isolation Phase B (CI Phase B) signal. 1-FCV-70-100-A is designed such that the periodic slave relay test will cause the valve to close, isolating cooling water to the RCP Oil Coolers. Failure of the valve to open after the test would lead to a unit shutdown. The other three valves are designed such that the slave relay test does not close the valves but utilizes continuity test circuits to verify slave relay operation and valve circuit integrity. This change will bring testing of 1-FCV-70-100-A in line with testing of the other three valves.

This change moves the CI Phase B signal for 1 -FCV-70-100-A to a different slave relay and adds a BLOCK test feature to the circuit. This will permit the slave relay test to be performed without isolating cooling water to the RCP Oil Coolers. The slave relays for these valves are tested every 92 days per Technical Specifications. This change will utilize existing abandoned cables previously used for the same function for a different containment isolation valve which was deleted. The cables have been evaluated and will be inspected for critical attributes to ensure they are acceptable for use in this application.

b. The Emergency Feeder Breakers (Emergency Diesel Generator Breakers) to the 6.9 KV Shutdown Boards are designed to trip on a Safety Injection (SI) signal when the Diesel Generator (DG) is operating in the parallel test mode. The slave relays which generate the SI signal for these breakers are tested every 92 days per Technical Specifications. Performing the slave relay test for this function at power would require declaring the DGs in the train being tested inoperable and entering the applicable LCO. The Technical Specifications also require testing of the DG breaker trip on SI every 18 months with the unit shut down to demonstrate test mode override. This ensures that DG availability during accident conditions will not be degraded as a result of testing.

This change moves the SL signal for the DG breakers to slave relays which are tested every 18 months, per the Technical Specifications, with the unit shut down. Justification for the 18 month frequency is provided in the TS Bases references for the existing functions of these relays. TS Change 95-090 will add a reference to DCN W-38238-B, which contains this SA/SE, in the TS Bases as the justification for this additional function. Performing the slave relay test for this function every 18 months will allow the test to be conducted coincident with the DG test mode override verification. This change will require relanding existing cables to different terminal blocks in the SSPS output cabinets. DCN W-38238-B provides restrictions for implementing the DG breaker circuit modification to ensure the TS requirements and FSAR commitments are met. The

Safety Evaluation Summary

This change does not affect any FSAR evaluations (accident analysis or equipment malfunction failures) previously performed. No new accidents or equipment malfunction failures are created. The Technical Specification is not affected. This change is in compliance with safety classification and performance requirements as specified in design basis documents. The performance of the DCN in stages does not compromise the safety of the plant. At the completion of each stage all affected components continue to meet their functional and safety requirements. Therefore, on the basis of the evaluation of effects, it can be concluded that the proposed change is acceptable from a nuclear safety standpoint and no unreviewed safety question exists.

modification for DG breakers can be implemented in operational modes 1 through 4 provided that only one of the four power trains (1A, 2A, 1B, or 2B) is removed from service at any one time. The power train removed from service must have the modification completed (including testing and MCR drawing update) and returned to service prior to beginning work on another power train. The modification can also be implemented in mode 5 provided that only one division of a power train (1A & 2A or 1B & 2B) is removed from service at any one time. The division of power train removed from service must have the modification completed (including testing and MCR drawing update) and returned to service prior to beginning work on the opposite division of power train.

SA-SE Number *WBPOSG-95-073-1*

Affected Documents	*Document Type*	*Safety Assessment Title*	*Implementation Date:*
DCN S-38520-A FSAR Change Pkg. 1402	DCN	N3-79-4001 Vendor Drawings and the FSAR Revised with the Latest Revision of Westinghouse CNFD Spec F5	8/27/96

Description of Change, Test, or Experiments

The change authorized by DCN S-38520 accounts for different weights of fuel and buoyancy effects encountered in the initial fuel load when the fuel must traverse between free air and water. The change also allows crane load limits to be set based on actual weights of the assemblies being manipulated not a fixed number. The changes are consistent with and based on Westinghouse fuel handling procedure F5 "Instructions, Precautions, and Limitations for Handling New and Partially Spent Fuel Assemblies," and Westinghouse letter subject "Refueling Machine Slack Cable Settings" dated November 11, 1995.

Safety Evaluation Summary

The conclusions reached in this Safety Evaluation is the change is safe and the change does not constitute an Unreviewed Safety Question. The basis for this conclusion is seated in the methodology used to determine manipulator crane settings. The old method of determining crane settings used fixed numbers of nominal weights and resistances. The new method uses individual fuel bundle weights . Thus the crane settings are tailored to individual conditions. This method still offers the same protective settings and therefore does not reduce nuclear safety or increase the possibility or probability of an accident nor reduce the margin of safety as defined by Tech Specs

SA-SE Number *WBPOSG-95-101-1*

Affected Documents	*Document Type*	*Safety Assessment Title*	*Implementation Date:*
DCN W-38579-A	DCN	Manual globe valve in the common discharge header for both heat exchanges of a diesel generator set	10/8/96

Description of Change, Test, or Experiments

This DCN will add an 8" manual globe valve (ASME Section III Class 3) in the 10" common discharge header for both heat exchanges of a diesel generator set. This will allow the discharge butterfly valves to be full open or almost full open and the additional globe valve to be throttled to absorb the excess system pressure and avoid the detrimental effects of cavitation at the butterfly valve.

The valve is an ASME Section III Class 3 valve and the piping has been analyzed for the additional loading. A piping support is added to the line to support the weight and to support the loads during a design basis earthquake.

This addition will not change the minimum flow requirement of the diesel generator heat exchanges which remains at 650 gpm per heat exchanger. The ERCW system at the diesel generator will have to be rebalanced using the globe valve for the majority of the throttling and final balancing using the outlet butterfly valves. The settings should not change between outages.

This DCN will be staged to allow incremental installation. Each diesel generator is a separate stage.

Safety Evaluation Summary

The system pressure drop has been evaluated and since there is excess head available the addition of this valve will not have any detrimental effects on the system flow or heat transfer capabilities. The addition of this valve will enhance the reliability of the ERCW supply to the diesel generators and will not introduce any unacceptable risk to the system. The installation has been designed to meet all requirements and will be installed as designed. The addition is acceptable with respect to nuclear safety.

SA-SE Number *WBPOSG-96-035-1*

Affected Documents

FSAR Change Pkg. 1415
DCN M38697-A
TRM Pkg 96-004

Document Type

DCN

Safety Assessment Title

Use of MDAFWP Pressure Switches to Detect Loss of CST

Implementation Date:

2/28/96

Description of Change, Test, or Experiments

The preferred water source for the Auxiliary Feedwater Pumps (AFW) pumps is the Condensate Storage Tank (CST). Since the CST is not a qualified source, Essential Raw Cooling Water (ERCW) serves as an alternate source when the CST is not available. A separate ERCW header is provided for each of the Motor-Driven AFW Pumps (MDAFWP) 1A-A & 1 B-B; the Turbine-Driven AFW Pump (TDAFWP) can receive water from either header. Two motor-operated valves (MOVs) are provided in each suction line to ERCW. Transfer to the ERCW is automatically initiated on low pressure in the AFW pumps suction lines. Loss of the CST is detected by pressure switches located in each of the pump suction lines from the CST.

During unit startup, with the MDAFWPs at low forward flow and the CST available, the TDAFWP suction was automatically transferred to ERCW. This spurious operation of the pressure switches was apparently caused by pressure oscillations in the CST header resulting from low flow operation of the MDAFWPs. DCN M-38697-A abandons the TDAFWP pressure switches and will use the MDAFWP pressure switches to detect loss of CST and initiate transfer of TDAFWP supply to ERCW.

Safety Evaluation Summary

DCN M-38697-A abandons the TDAFWP pressure switches and will use the MDAFWP pressure switches to detect loss of CST and initiate transfer of TDAFWP supply to ERCW. Redundancy and electrical separation will be maintained. The setpoints of the pressure switches do not change but the time delay for the transfer does. The overall AFW injection time of 60 seconds remains unchanged. The PRA evaluation indicates that there is no increase in failure probability. Therefore, this change does not affect any FSAR evaluations of accident analysis or equipment malfunction failures previously performed. The change does not alter the function of the AFW pumps to remove reactor core decay heat and RCS stored heat. The change will maintain the automatic transfer to the emergency water supply as required while avoiding spurious transfers when the CST is available. The time delay for the CST transfer to ERCW will be increased to allow additional time for riding through possible perturbations in the AFW pumps suction headers, thus decreasing the possibility of spurious transfers potentially injecting ERCW into the SGs and secondary plant systems instead of the preferred source (CST).

The time delay relay settings were also analyzed and shown to be adequate to meet response time and accuracy requirements, and to assure that the AFW pumps still have adequate NPSH. The modification does not introduce any additional components or components of a different type which could create a different failure mode. Reliability of the AFW system components will not be degraded as a result of this change. This change is in compliance with safety classification requirements as specified in design basis documents. Therefore, on the basis of the evaluation of effects, it can be concluded that the proposed change is acceptable from a nuclear safety standpoint and no unreviewed safety question exists.

SA-SE Number *WBPOSG-96-056*

Affected Documents	*Document Type*	*Safety Assessment Title*	*Implementation Date:*
DCN W-38598-B	DCN	Additional Sampling Points Added to Provide Grab Sample Capabilities	2/13/97

Description of Change, Test, or Experiments

Summary of the proposed change, test or experiment and SAR impact. DCN W-38595-B will add two sampling points which will be used for sampling during the time when a.) the waste gas analyzer is out of service or b.) when valve O-PCV-77-103 is out of service. Each item, a. or b., will have a parallel flow paw to an isolation valve and terminate in a self sealing quick disconnect to mate with the quick disconnect on the sample bottles.

The first point is a bypass around the Waste Gas Analyzer found in the Unit 2 hot sample room. The DCN will 'Tee' into the 3/8" inlet line to the Waste Gas Analyzer and provide a Sample path to the grab sample bottle station located on the wall in the hot sample room. The line will be identical to components already installed with a check valve and an isolation valve terminating ire a quick disconnect. This will allow an alternate grab sample point in the event the Waste Gas Analyzer is unavailable for any reason. The current design is such that several sources can be sampled at the Waste Gas Analyzer, this setup utilizes an exhaust line to enable purging the lines to control the sample quality. This change will allow the new sampling point to use the exist exhaust line for purging.

The second point is to add a bypass sampling point around valve O-PCV-77-103. This pressure control valve regulates the pressure in the decay tank waste gas sample line to the normal sampling location located at the Waste Gas Analyzer in the U2 hot sample room. The valve is located in the Waste Gas Decay Tank gallery in the Auxiliary Building. The tubing is routed close to the floor, along the wall and parallel to the existing header. This will bring the grab sample point close to the access door but still within the gallery. The total added length is approximately 15'. The tap into the existing line will be a welded joint and the remaining joints will be compression fittings. This addition will allow a sample to be taken from the waste gas decay tanks in the event the O-PCV-77-103 valve is out of service for any reason. The ALARA programs will not be adversely affected as the normal accessible areas will not be impacted.

Revision B of this DCN establishes a piping class break at the isolation valve O-ISV-043-0233, which is the manual isolation valve of the new sampling point added by this DCN upstream of O-PCV-77-103. The valve and tubing upstream of the isolation valve is class D; downstream of this valve will be changed to class G piping (approximately 8" of tubing, a quick disconnect, and a tubing cap). This class break will rematch the class break at the existing sampling point installation (0-ISV-43-774 and -777) at the waste gas analyzer in the U2 hot sample room.

Additionally, this revision provides for staging the work for the two sampling points. This will allow the work to progress in a more logical manner and allow the two pieces of work to be tagged out and closed independently. This will be significant in resuming the affected sections to service. The work associated with the sampling point around 0-PCV-77-103 is stage 1 and the work associated with the analyzer sampling point is stage 2.

Safety Evaluation Summary

The changes to the system as provided for in this DCN will provide alternative sampling paths to existing lines in the event of equipment failure. Should the Waste Gas Analyzer fail or the pressure control valve (0-PCV-77-103) fail, this modification will allow continued operation by providing alternative means to sample the waste gas. The new sampling points are not required by the current text in the SAR nor in the Tech Specs but will allow continued operation by meeting Tech Spec requirements on sampling during operation Although new components are being added to the system, they are designed and will be installed using the appropriate criteria to match the existing installation of grab sample points. The probability of an inadvertent release beyond the limits of 10CFR20 and 10CFRI00 is not increased. The new configuration and components will be acceptable with regard to nuclear safety.

SA-SE Number *WBPOSG-96-089-1*

Affected Documents

TACF NO. 1-96-19-006

Document Type

Temporary Alteration

Safety Assessment Title

Safety evaluation is provided to support plant operation above 85% power with TACF 1-96-19-006 in effect.

Implementation Date:

4/23/96

Description of Change, Test, or Experiments

Revision 1 of SA/SE WBPOSG-96-089 supports plant operation with TACF 1-96-019-006 in place beyond the 85% power level up to 100% power. TACF 1-96-019-006 disables the #3 heater drain tank pump cavitation protection circuit on 1-LCV-006-0106A. The current design of the #3 heater drain tank pump cavitation protection circuit drives the 1-LCV-6-106A valve to a pre-determined mass flow (equates to a approximate 30% valve open position) upon receipt of a high differential pressure signal across PDIS-6-106A, -106B, or -106C. The TACF is necessary because of unnecessary automatic closures of 1-LCV-6-106A to its 30% open position which results in high heater drain tank levels which result in turbine runbacks and unnecessary plant transients.

During power ascension testing, start-up strainers have been used to aid in secondary side clean-up. This SA/SE will contain a special requirement to remove the start-up strainers prior to exceeding the 85% power level. This special requirement is for the purpose of eliminating the possibility of loss of a #3 heater drain tank pump due to a clogged filter.

FSAR Impact: FSAR Section 10.4.10.3 discusses how at power levels greater than 85%, a trip of a #3 heater drain pump produces a high differential pressure between the #3 heater drain tank and the pump suction and initiates a protection function to reduce flow on the remaining pumps to protect against insufficient NPSH. The protective circuit exists to prevent the two remaining operable pumps from running out trying to respond to heater drain tank level control valve position. This could potentially cause the pumps to experience loss of NPSH and is the condition which the disabled circuit is intended to protect against. If more than one #3 heater drain tank pump trips out above the 85% power level, then the main feedwater system may not be capable of maintaining level in the steam generators, and could result in a reactor trip. However, this event would be bounded by Loss of Normal Feedwater Event (Ref: FSAR Section 10.2.8). Therefore, compensatory measures are required by this SE to ensure that the above described situation does not occur during the short time in which TACF 1-96-019-006 is in effect. The compensatory measures will assure that the overall protective function of the disabled circuit will be performed manually to assure that unnecessary reactor trips do not impact the allowable lifetime accumulative design transients for this type of event, while the TACF is in place.

Safety Evaluation Summary

TACF 1-96-019-006 does not affect any SAR evaluations (accident analysis or equipment malfunctions) previously performed and is bounded by the Loss of Normal Feedwater Event in FSAR Section 15.2.8. No new accidents or equipment malfunction failures are created. The Technical Specification (and TRM) is not affected by the implementation of this TACF. Therefore, on the basis of the evaluation of effects, it is concluded that the proposed change is acceptable from a safety standpoint and no unreviewed safety question exists.

SA-SE Number *WBPTSS-97-047-0*

Affected Documents

FSAR Change Pkg. 1467

Document Type

FSAR

Safety Assessment Title

Corrections for two tables in FSAR

Implementation Date:

2/11/97

Description of Change, Test, or Experiments

The change being considered is FSAR Change Package 1467. This change package makes changes in two tables in the FSAR in response to two PERs documenting discrepancies in the tables. The following is a list of the changes being made for which a safety evaluation is necessary:

1. Table 3.9.17, sheet 5 of 8, is changed by modifying the Function/Description column for valves CKV-63-502 and CKV-63-510 from "ECCS Flowpath Integrity/RCS Press. Bound. Prot." to "ECCS Flowpath Integrity"

RCS pressure boundary isolation valves are required to have their seat leakage integrity [leak rate] determined by testing in accordance with TS 3.4.14. Although valves CKV-63-502 and CKV-63-510 are in reality not part of the Reactor Coolant System Pressure Boundary, the identification of them as providing an RCS Pressure Boundary Protection function in table 3.9-17 implies a requirement to quantify the seat leakage of the valves by testing in accordance with the TS. Removal of this reference alters this perception.

2. Table 3.9-26, sheet 2 of 4, is changed by deleting reference to valves FCV-67-22, FCV-67-81, FCV-67-82, FCV-67-147, and FCV-70-153. These are valves that have been locked in their safe position with power removed from their actuators in response to the considerations of 10CFR50, Appendix R.

Category S-Passive valves are required to have their remote position indicators verified by testing once per two years. Category B-Active valves are also required to be stroke time tested. Although valves FCV-67-22, FCV~7-81, FCV-67-82, FCV-67-147 and FCV-70-153 have in reality been locked in their safe position with power removed from their actuators, thus defeating their remote position indication as well as their ability to be remotely actuated for stroke time testing, the identification of the valves in FSAR table 3.9-26 as Category B implies that these valves are tested. Removal of the valves from this table alters this perception.

Safety Evaluation Summary

10CFR50.2 provides definitions of terms including the reactor coolant pressure boundary as being, "All those pressure containing components of boiling and pressurized water cooled nuclear power reactors, such as pressure vessels, piping, pumps and valves which are.... (ii) The second of two valves normally closed during normal reactor operation in system piping which does not penetrate primary reactor containment.

The provisions of this paragraph stop the Reactor Coolant Pressure Boundary at the second normally closed valve during normal reactor operation. Valve CKV-63-502 is the fourth normally closed valve in the SIS cold leg injection flow path, and the fifth normally closed valve in the SIS hot leg injection flow path. In addition, there are three normally open Motor Operated Valves [MOVs] in the SIS cold leg injection flow path and one normally open MOV in the SIS hot leg injection now path that can be closed by operator action to isolate CKV-63-502 from the reactor. Valve CKV-6i3-510 is the fifth normally closed valve in the RHR cold leg injection flow path and the sixth normally closed valve in the RHR hot leg injection flow path. In addition, there are two additional MOVs and one additional Air Operated Valve [AOV) in both the RHR cold and hot leg injection How paths that can be closed by operator action to isolate CKV-63-510 from the reactor.

10CFR50.55a(c)(1) requires components which are part of the reactor coolant pressure boundary to meet the requirements for Class 1 components of Section III of the ASME Boiler and Pressure Vessel Code. Valves CKV-63-502 and CKV-63-510 are both Code Class 2 valves and are installed in Code Class 2 piping. Therefore they are not designed or installed to the requirements of the reactor coolant pressure boundary.

Valves CKV-63-502 and CKV-63-510 should not be considered part of the reactor coolant pressure boundary because they are not designed to the requirements of the reactor coolant pressure boundary and are not required to be considered part of the reactor coolant pressure boundary. If they are not part of the reactor coolant pressure boundary, they cannot provide a pressure isolation function to protect. the reactor coolant pressure boundary. Therefore it is improper to make the statement that they provide a reactor coolant pressure boundary protection function. The statement to this fact in the FSAR is in error and removal of the statement does not create an adverse or unsafe condition.

The statement that the valves provide an ECCS flow path integrity function is correct and is reflected in the testing requirements imposed by the IST Program [SSP-8.06, Attachment 1], which categorizes both valves as C-Active and imposes a requirement to full stroke each valve to both the open and the closed positions.

Valve FCV-67-147 was originally intended to be an active valve. As originally conceived, ERCW would normally operate with the cooling water to the A ERCW/CCS heat exchanger, which is Train A. being supplied by the Train B ERCW pumps. Upon loss of Train B power, several valves, including FCV-67-147, would change position in order to automatically realign the A E.RCW/CCS heat exchanger to be supplied with cooling water from the Train A ERCW pumps. For unit 1 operation, this concept has been dropped. The valves originally required to change position to align the A ERCW/CCS heat exchanger to be fed from the Train A ERCW pumps have been placed in the position necessary to use the Train A pumps all the time. Therefore there is no need for these valves to change position. Specifically, valve FCV-67-14, has been locked in the position necessary to supply the train A ERCW/CCS heat exchanger with the Train A ERCW pumps with power removed from its actuator. Since the valve is no longer required to change position, it is not required to be stroke time tested. Additionally, with power removed from the actuator the position indicators are not functional. Therefore. verification of the accuracy of the remote position indicators is also not required. For these reasons, removing this valve from this table, and thereby removing the implication that it is tested, does not create an adverse or unsafe condition.

Valve FCV-70-153 was originally intended to be an active, category B valve. As originally conceived, Component Cooling System (CCS) water flow through the Residual Heat Removal (RHR) system train B heat exchanger would normally be isolated by FCV-70-153. When cooling water flow was needed, the valve would be opened by operator action to provide the necessary flow. However, for unit 1 operation, this plan has been changed. During the operation of unit 1 only, valve 1-FCV-70-153 has been locked in the open position with power removed from its actuator. Thus cooling water flow is always present on the unit 1 train B RHR heat exchanger and the unit 2 train B RHR heat exchanger is prohibited from robbing flow. Since the valves are no longer required to change position, they are not required to be stroke time tested.

Additionally, with power removed from their actuator, the position indicators are not functional. Therefore, verification of the accuracy of the remote position indicators is also not required. For these reasons, removing these valves from this table, and thereby removing the implication that they are tested, does not create an adverse or unsafe condition.

Valves FCV-67-22, FCV-67-81, and FCV-47-82 are normally open valves that remain open during all creditable accident conditions. In response to the 10CFR50, Appendix R, these valves have been locked in the open position with power removed from their actuator. Since the valves are not required to change position, they are not required to be stroke time tested. Additionally, with power removed from the actuator, the position indicators are not functional. Therefore, verification of the accuracy of the remote position indicators is also not required. For these reasons, removing these valves from this table, and thereby removing the implication they are tested, does not create an adverse or unsafe condition.

SA-SE Number *WBPTSS-97-158-0*

Affected Documents	*Document Type*	*Safety Assessment Title*	*Implementation Date:*
FSAR Change Pkg. 1482	FSAR	Relief Valves Assigned to the Augmented Inservice Testing Program	7/29/97

Description of Change, Test, or Experiments

Certain relief valves were originally assigned to the Augmented Inservice Testing Program. These valves had been considered as providing only a thermal expansion protection function to protect shutdown or out of service equipment. The criteria for inclusion in the ASME XI Inservice Testing Program is that a valve must be required to fulfill a specific purpose in order to: [1] mitigate the consequences of a reactor accident, [2] achieve the cold shutdown condition, or [3] maintain the cold shutdown condition. Since the valve's apparent function did not meet any of these criteria, the valves were placed in the Augmented Inservice Testing Program. The requirements for testing in the Augmented Inservice Testing Program are identical to the requirements for testing in the ASME XI Inservice Testing Program, except that a documented pre-service test is not required of valves in the Augmented Inservice Testing Program.

It was subsequently determined that these valves provided overpressure protection functions that should have caused the valves to be included in the ASME XI Inservice Testing Program. These functions were either to protect a low pressure, safety related component within the associated system or to protect against overpressure protection due to a tube rupture in the associated heat exchanger. WBPER970117 was written to document this condition.

Valve work histories were reviewed to confirm that appropriate pre-service testing had in fact been accomplished for these valves. The ASME XI and Augmented Inservice Testing Programs have been revised to list the testing requirements for the valves in the correct program. The implementing procedure for valve testing has been revised to require the future testing of the valves to be in accordance with the ASME XI Inservice Testing Program. FSAR Change Package 1482 revises SAR table 3.9-26 to include the twelve valves affected by this condition. This Safety Assessment/Evaluation addresses FSAR Change Package 1482.

Safety Evaluation Summary

This proposed change does not alter the Inservice testing requirements for the subject valves. Both the Augmented and the ASME XI Inservice Testing Programs mandate testing of safety and relief valves in accordance with ASME OM-1. The only difference in the Augmented and ASME Section XI Inservice Testing Programs with regard to safety and relief valves is that the Augmented Program does not require a documented pre-service test. Since WBPER970117 determined that an acceptable pre-service test was in fact performed, the subject valves meet all requirements of ASME Section XI for inclusion in the ASME XI Inservice Test Program. Relocating the requirement to perform Inservice testing of a safety valve to a higher tier program does not create an adverse or unsafe condition.

SA-SE Number *WBRAD-96-008-0*

Affected Documents	*Document Type*	*Safety Assessment Title*	*Implementation Date:*
TACF No. 0-96-54-77 SOI-77.01	Temporary Alteration	TACF allows a check valve and manual isolation valve to be installed at the CDCT filter vant and monitor tank pump casing drain outlets.	11/3/97

Description of Change, Test, or Experiments

This TACF allows a temporary check valve and manual isolation valve to be installed at the Cask Decon Collector Tank (CDCT) filter (O-FLTR-77-IA) vent and the Monitor Tank pump (O-PMP-77-2906) casing drain piping connections. A flush is required to allow the background setpoint for the release monitor (O-RE-90-122) to remain below the maximum allowed by plant procedures. The CDCT vent pipe cap and the Monitor Tank pump casing drain blind flange will be temporarily removed to facilitate connection of the valves. Installation of these valves will allow the process piping from the CDCT and Monitor Tank through the discharge liquid radwaste release monitor to be flushed with clean water after each liquid release. The Waste Disposal System design or functional requirements presented in the FSAR are not impacted.

Safety Evaluation Summary

This activity is safe because the check valve and manual isolation valves used will be rated to withstand the expected pressure of the system when aligned for normal flushing activities. TACF 0-96-54-77 adds temporary valves that enable the operators to flush the liquid process piping after a release has been made. It does not change the system design basis, logic or function of any system that is important to safety. This TACF does not increase the probability of occurrence of an accident. Nor does it increase the probability of occurrence of a malfunction of equipment important to safety. There are no credible failure modes that would cause the Liquid Waste Disposal System to be unable to perform its function of processing water. Nor does this change affect any equipment failure modes that are important to safety. This system is not used for any accident mitigation function. The addition of the valves will not prevent any component from performing its function as described in the Technical Specifications.